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RADIOLOGICAL SURVEYS OF IDAHO PHOSPHATE ORE PROCESSING—THE THERMAL PROCESS PLANT

NOVEMBER 1977

U.S. ENVIRONMENTAL PROTECTION AGENCY
OFFICE OF RADIATION PROGRAMS
LAS VEGAS FACILITY
LAS VEGAS, NEVADA 89114

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16. ABSTRACT Radiological surveys conducted at the FMC Corporation's Thermal Process Plant in Pocatello, Idaho indicate slightly elevated ambient levels of natural radioactivity within the plant. Compared to an estimated natural background annual dose equivalent rate of about 79 mrem, net gamma exposure rates ranged from 42 mrem in general plant areas to 182 mrem per work year on the slag pile. Ambient radon-222 concentrations, ranging from 0.17 to 1.4 pCi/l, were measured in several indoor locations, but 11 pCi/l was measured in the Control Room of the Condenser and Fluid Bed Building. Elevated airborne radioactivity concentrations, orders of magnitude greater than measured background concentrations, were also measured in several work areas, with polonium-210 and radium-226 being the most predominant radionuclides of the natural uranium decay series. Particle size characterization indicates roughly 50 percent of the arithmetic total radioactivity is associated with the particle size fraction less than one micrometer equivalent aerodynamic diameter. Stack sampling results also show that appreciable concentrations of the naturally-occurring radionuclides, particularly polonium-210 and uranium, are being discharged into the local environs. A general radioactivity balance indicates that the ore is the source of essentially all of the inlet radioactivity. The slag accounts for essentially all of the outgoing uranium and radium-226 and up to about 50 percent of the lead-210 and polonium-210. The stack effluents account for about 20 percent of the total polonium-210 activity.

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This report has been reviewed by the Office of Radiation Programs - Las Vegas Facility, U. S. Environmental Protection Agency, and approved for publication. Mention of trade names or commercial products does not constitute endorsement or recommendation for their use.

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The field conduct of this study was directed by Joseph A. Cochran, whose present address is:

Chief - Sanitation Branch
Directorate of Facilities Engineering
U.S. Department of the Army
Fort Ord, California 93941

This report has been prepared by:

Gregory G. Eadie and David E. Bernhardt
Office of Radiation Programs-Las Vegas Facility
U.S. Environmental Protection Agency
Las Vegas, Nevada 89114

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PREFACE

The Office of Radiation Programs of the U.S. Environmental Protection Agency carries out a national program designed to evaluate population exposure to ionizing and nonionizing radiation, and to promote development of controls necessary to protect the public health and safety. This report describes various surveys which were conducted at the FMC Corporation's Thermal Process Plant near Pocatello, Idaho to provide basic data to be used for the evaluation of the radiological impact associated with the phosphate industry. Readers of this report are encouraged to inform the Office of Radiation Programs of any omissions or errors. Comments or requests for further information are also invited.

Donald W Hendricks

Donald W. Hendricks
Director, Office of
Radiation Programs, LVF

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ABSTRACT

Radiological surveys conducted at the FMC Corporation's Thermal Process Plant in Pocatello, Idaho indicate slightly elevated ambient levels of natural radioactivity within the plant. Compared to an estimated natural background annual dose equivalent rate of about 79 mrem, net gamma exposure rates ranged from 42 mrem in general plant areas to 182 mrem per work year on the slag pile. Ambient radon-222 concentrations, ranging from 0.17 to 1.4 pCi/l, were measured in several indoor locations, but 11 pCi/l was measured in the Control Room of the Condenser and Fluid Bed Building.

Elevated airborne radioactivity concentrations, orders of magnitude greater than measured background concentrations, were also measured in several work areas, with polonium-210 and radium-226 being the most predominant radionuclides of the natural uranium decay series. Particle size characterization indicates roughly 50 percent of the arithmetic total radioactivity is associated with the particle size fraction less than one micrometer equivalent aerodynamic diameter. Stack sampling results also show that appreciable concentrations of the naturally-occurring radionuclides, particularly polonium-210 and uranium, are being discharged into the local environs.

A general radioactivity balance indicates that the ore is the source of essentially all of the inlet radioactivity. The slag accounts for essentially all of the outgoing uranium and radium-226 and up to about 50 percent of the lead-210 and polonium-210. The stack effluents account for about 20 percent of the total polonium-210 activity.

INTRODUCTION

Recent reports (Windham et al., 1976; Guimond and Windham, 1975) indicate that mining and processing of phosphate ores technologically enhance the quantities of naturally-occurring radioactive materials such as radium and uranium available to man in the environment. Such radionuclides have relatively long half-lives and will therefore persist in man's habitat for thousands of years.

Current estimates show that the phosphate industry has generated about twenty times the volume of waste materials as compared to that from the uranium mining and milling industry. The total radioactive waste inventory generated by the uranium and the phosphate mining and milling industries are comparable to each other although the per gram radioactivity content of the phosphate waste is much less than that of uranium waste material (EPA, February 1977).

In general, the phosphate industry is neither regulated nor monitored for the possession, use, or discharge of radioactive materials associated with phosphate rock and its products and by-products. Recently, the State of Idaho (June 1, 1977) has prohibited the use of phosphate slag material in the construction of habitable structures, but has permitted the continued use of slag for road construction, railroad ballast, and other general purposes.

Of immediate concern is the accumulation of a data base which will ultimately lead to an assessment of the impact on public health due to the phosphate industry's activities. This report discusses various radiological surveys conducted in the FMC Corporation's Thermal Process Plant in Pocatello, Idaho. A similar study, conducted in a wet-process plant, will be discussed in another report.

THE THERMAL PROCESS

Phosphate rock is obtained from the Gay Mine located 35 miles northeast of the Thermal Process Plant in Pocatello, Idaho. At the plant, phosphate rock is crushed, screened, and briquetted. The briquettes are then fed into calciners where they are heated to 2400° F. This calcining serves two purposes: 1) to burn out organic material and 2) to heat-harden agglomerates so they will withstand further processing steps without disaggregating. The calcined nodules pass through a proportioning building where sized coke and silica are blended in. This mixture is called the "burden" and flows by conveyor belt into the electric furnaces. In the furnace, the high temperature reaction (2500° to 8000° F) drives off two gases - phosphorus and carbon monoxide. It leaves two molten residues - slag and ferrophosphorus (a mixture of iron, vanadium, chromium, etc.). A simplified chemical equation for the electric furnace reaction is:

$$2Ca_3(PO_4)_2 + 6SiO_2 + 10C = P_4 + 10CO + 6CaSiO_3$$

Off-gases are treated in electrostatic precipitators for dust removal, then in a waterspray cooler where the gaseous elemental phosphorus is condensed, collected in a sump, and pumped to storage. Most of the carbon monoxide is used in the calciner but some is burned to become carbon dioxide, which is released to the atmosphere via the flare stack.

The waste product slag, which is predominately calcium silicate, is crushed and used as an aggregate in highway surfacing material and as railroad track ballast. Some other uses of slag have been for concrete aggregate material and for use in

insulation products. The ferrophosphorus residue is usually custom crushed and processed to extract vanadium at another plant. The finished product, elemental phosphorus, is shipped to other plants for use in detergents, cleaning products, and in numerous other applications.

In brief, the FMC Corporation's Thermal Process Plant handles over 2,000,000 tons of raw materials (1,800,000 tons of phosphate rock; 190,000 tons of coke; and 130,000 tons of silica) and produces 125,000 tons of elemental phosphorus per year. From a material balance viewpoint, about 2,000,000 tons of waste by-products (i.e., slag and ferrophosphorus) are generated each year. A simplified flow diagram of this thermal process is shown in Figure 1.

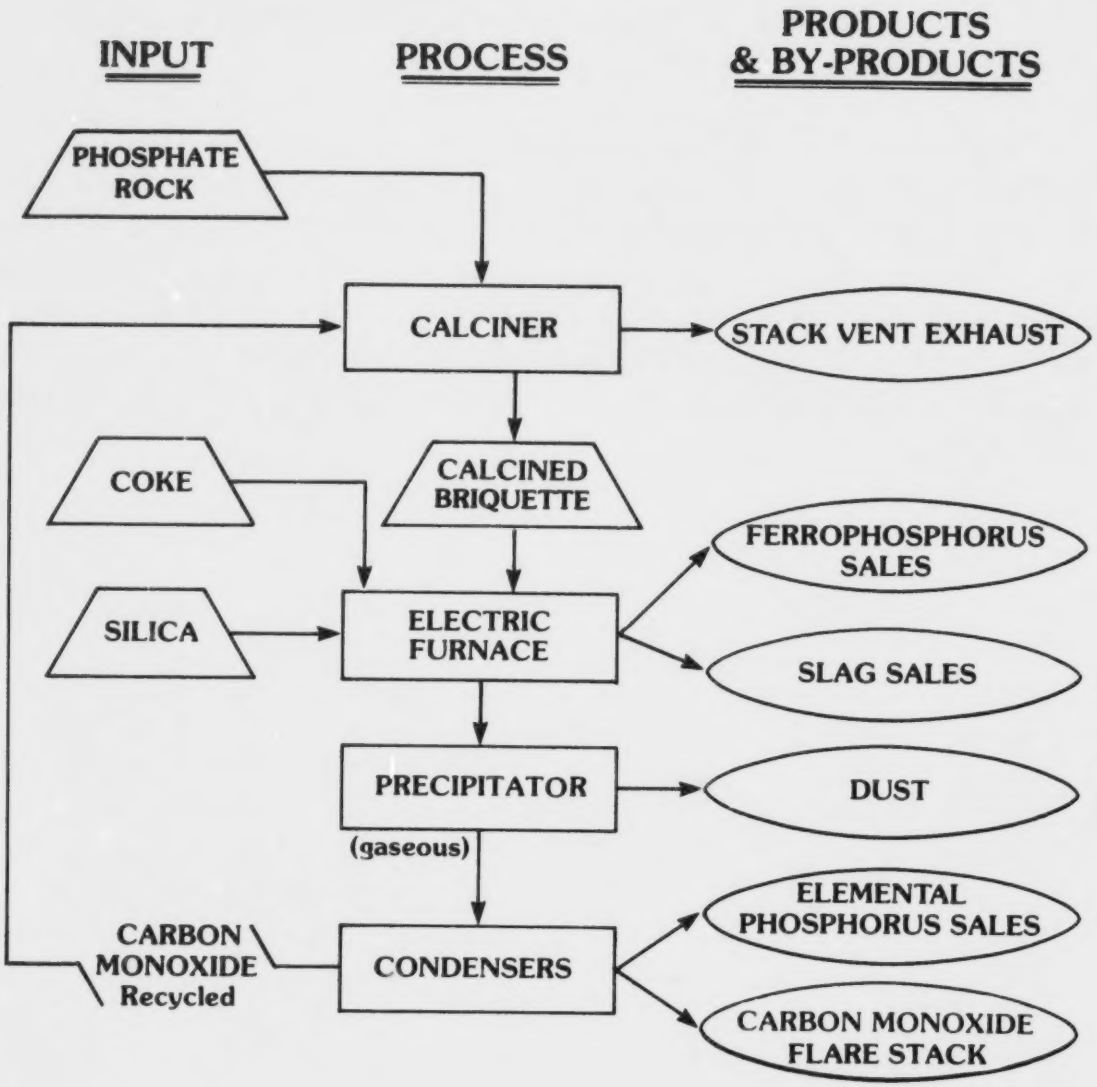

Figure 1. Flow diagram of the thermal process

THE RADIOLOGICAL ASSESSMENT OF THE THERMAL PROCESS PLANT

Radiological surveys were completed at the FMC Corporation's Thermal Process Plant in Pocatello, Idaho during the week of May 5 to 8, 1975. Included among these surveys were: gamma radiation measurements; ambient radon-222 concentrations; radiological analyses of selected input, process and product samples; airborne particulate sampling; and particle-size characterization. Stack sampling of representative effluent discharge vents was also completed in September 1975. The following sections of this report discuss the results of these surveys.

GAMMA RADIATION SURVEYS

In order to assess the gamma radiation exposure rates in the various working areas of the Thermal Process Plant, a survey was conducted using a portable gamma scintillator survey meter.* This instrument was calibrated with a radium-226 standard, and measured the relative gamma radiation exposure rate in units of microroentgens per hour (μR/h). Measurements were made at a height of three feet above an area surface and the results shown in Table 1 represent average values for each location. At the time of these surveys, a pressurized ionization chamber (PIC) was not available to complete radiation surveys for comparison to the portable scintillator measurements. Although the scintillator survey meter's response is dependent on the energy of the gamma photon (Eadie et al., 1976), no instrument response correction factor has been applied to the reported scintillator measurements. Therefore, for the following dose estimates, the reported scintillator measurements (Table 1) are assumed to be equivalent to the "true" gamma exposure rate due to both the cosmic and the terrestrial components.

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* Baird-Atomic, Type NE148A-Gamma Scintillator Ratemeter.

TABLE 1. GAMMA RADIATION SURVEYS[+]
GENERAL WORK AREAS

LOCATION DESCRIPTION	GROSS GAMMA EXPOSURE RATE μR/h	NET GAMMA EXPOSURE RATE* μR/h	ESTIMATED WORK YEAR DOSE EQUIVALENT** mrem/year
Shale Area: lunch room	25	16	32
Ore Dump Unloader Area:			
Outside	70	61	122
Inside Control Room	20	11	22
Coke Dryer: Outside	60	51	102
Inside Control Room	28	19	38
Pelletizer Building:			
General Floor Area	15	6	12
Calciner #1:			
Loading Area	20	11	22
Inside Control Room	12	3	6
Proportioning Building:			
General Floor Area	15	6	12
Inside Lower Control Room	40	31	62
Furnace Building:			
West End, Lower Level -			
General Floor Area	10	1	2
Control Room #3-4 (inside & outside)	8	BKG.	BKG.
Foreman's Office	20	11	22
Fluid Bed Control Rm.	30	21	42
General Area Near Furnace #4	30	21	42
West End, Near Slag Run	35	26	52

TABLE 1. GAMMA RADIATION SURVEYS+
GENERAL WORK AREAS(Cont'd)

LOCATION DESCRIPTION	GROSS GAMMA EXPOSURE RATE μR/h	NET GAMMA EXPOSURE RATE* μR/h	ESTIMATED WORK YEAR DOSE EQUIVALENT** mrem/year
Ferrophosphorus			
Cooling Area #2	20	11	22
Cooling Area #3	16	7	14
Slag Pit behind Building	30	21	42
Slag Area #1	25	16	32
Slag Area:			
General Outside Area	100	91	182
Technical Building:			
Inside	10	1	2
Outside	20 to 40	11 to 31	22 to 62
Old Kiln Building:			
Inside	15	6	12
General Plant Areas:			
Outside roadway between Old Kiln Bldg. and Proportioning Bldg.	60	51	102
Outside roadway between Furnace Bldg. and Pelletizer Bldg.	60	51	102
Background Radiation			
General Pocatello, Idaho Area	9	-	18 mrem per Work Year 79 mrem per Year for continuous exposure

+These surveys were conducted on May 8, 1975 using the Baird-Atomic NE148A-Gamma Scintillator Ratemeter. All measurements are average instrument response, in units of μR/h, at three-foot height above surface area.

*An estimated background gamma exposure rate of 9 μR/h (using the scintillator survey meter) has been subtracted from the gross value to obtain the net result. (This instrument does not respond fully to cosmic radiation. Thus, neither the estimated background nor the gross readings reflect the true contribution of the cosmic component.)

**Work Year has been defined as 40 h/wk for 50 wk/y (i.e., 2000 hours). The Work Year Dose Equivalent is the net value multiplied by the Work Year time equivalent;i.e., the dose estimate due to continuous work year occupancy per specific area. This may be an overestimate, depending on the actual occupancy factors.

EXTERNAL DOSE ASSESSMENT

Background Radiation

Gamma exposure rate data for non-coastal plain regions of the United States have been summarized and indicate a mean value of 5.2 μrem/h (46 mrem/year) absorbed dose rate equivalent in air from terrestrial sources (NCRP, November 1975). The absorbed dose rates (mrad/year) have been converted to dose equivalents (mrem/year) using a quality factor of one for gamma photons, which is consistent with the recommendations of the National Council on Radiation Protection (NCRP, January 1971). Specific surveys around the National Reactor Testing Station near Idaho Falls, Idaho showed an average terrestrial dose equivalent rate of about 6 μrem/h (Oakley, 1972). Therefore, the natural terrestrial background radiation annual dose equivalent for the Pocatello, Idaho area should be approximately 53 mrem per year (i.e., continuous exposure at 6 μrem/h x 24 h/day x 365 day/year). The average dose equivalent rate due to cosmic radiation in the Pocatello area has been estimated to be roughly 3 μrem/h, for a scintillator survey meter measurement. This value corresponds to an annual average cosmic dose equivalent rate of 28 mrem per year as summarized by NCRP (November, 1975). Therefore, a total natural background radiation exposure rate of 9 μrem/h has been used to obtain the "net" exposure rates for the survey results as reported in Table 1.

In-Plant Radiation Exposures

The lowest exposure rate (8 μR/h) in the Thermal Process Plant was essentially the background exposure rate and was measured in the Control Room #3-4 in the Furnace Building (Table 1). Therefore, for workers who spend the majority of their working periods inside this control room, the maximum annual dose equivalent would be about the average background rate of 18 mrem per year (i.e., 9 μR/h x 40 h/wk x 50 wk/year). General working

areas within the Furnace Building had net exposure rates ranging from 1 μR/h in the West End, Lower Level to 26 μR/h around the slag run. For general working areas of the Furnace Building, the average net exposure rate was 21 μR/h, which corresponds to a work year dose equivalent of 42 mrem/year.

For the general working areas of the Calciner, the net exposure rate was 11 μR/h. Working areas within the Pelletizer, Old Kiln, and the Proportioning Buildings had average net exposure rates of 6 μR/h. The average net exposure rate was 1 μR/h inside the Technical Building which also is essentially the background exposure rate.

The highest net exposure rate for any of the indoor working areas within the Thermal Plant was 31 μR/h in the Lower Control Room of the Proportioning Building. Therefore, the maximum work year dose equivalent for a worker in this area would be about 62 mrem per year.

In-plant radiation exposures are summarized in Table 1. By assuming continuous exposure during the entire work period (40 h/wk x 50 wk/year) at the net gamma exposure rate levels, the reported dose equivalents are probably overestimated. A more precise dose estimate could be obtained by fractioning the time spent in various work locations and using that particular exposure rate; however, such information has not been obtained in this study. It should also be emphasized that these in-plant external gamma dose estimates for the work year are in addition to the dose received by an individual due to natural background radiation exposure. Furthermore, these estimates are based on free air measurements. It is estimated that the dose to internal body organs varies from 0.5 to about 0.7 of the free air measurement due to the shielding effect of the body (O'Brien and Sanna, 1976).

Due to the use of waste-product slag for on-site roadway construction purposes, high net gamma exposure rates (averaging about 51 μR/h) were obtained out-of-doors on the roadways. Surveys around the ore dump unloader area indicate net exposure rates of about 61 μR/h. The highest net gamma exposure rate (91 μR/h) for any area in the Thermal Process Plant was out-of-doors in the slag dump area. Although work year dose equivalent rates have been shown in Table 1 for the out-of-doors areas, it is obvious that these values are overestimates since no worker spends his entire work shift directly on the slag pile or in the ore-storage area.

Surveys conducted at a Thermal Process Plant in Florida (Windham et al., 1976) showed external gamma dose rates slightly higher than those measured in this study. Total measured dose equivalent rates (including background) in the Florida plant ranged from 100 mrem per work year in general plant areas to a maximum of 300 mrem per work year in the ferrophosphorus and slag storage area. For the Idaho plant, the general work areas averaged about 42 mrem per work year, and the maximum net dose equivalent rate (on the slag pile) was 182 mrem per work year, above a background level of about 79 mrem per year.

AMBIENT RADON-222 CONCENTRATIONS

Sampling System and Analytical Methods

A continuous, low-volume sampling system was used to obtain samples of ambient air for analysis of radon content (U.S. Public Health Service, 1969). This sampling technique consists of drawing filtered air through a small, low-volume air pump (less than 10 ml/min sampling rate) into a 30-liter Mylar bag. The air intake was about one meter above the ground surface. Various sampling time periods were used for this study.

Radon analysis was completed at the Environmental Monitoring and Support Laboratory in Las Vegas, Nevada (EMSL-LV) using a radon concentration apparatus for sample preparation (Johns, 1975). This apparatus permits the isolation of radon from about a 5-liter ambient air sample by transferring the ambient air from the Mylar bag into a container of known volume, followed by circulation through water and carbon dioxide traps. Radon is collected on two charcoal traps maintained at a temperature of about minus 80° C using dry ice and acetone. The radon sample is then de-emanated into a Lucas scintillation cell (125-ml volume) using helium gas at 400° C. The Lucas cell is held for four and one-half hours to allow for the ingrowth of the radon daughters and then counted in a photomultiplier tube/scaler unit. The radon activity is calculated to the mid-point of the sample collection period to account for the radioactive decay ($T_{\frac{1}{2}}$ = 3.82 days) of radon from the time of sample collection to time of analysis. The ambient air volume sampled is determined by correcting for air density differences based on the nearest 1000-foot elevation increment, and the average ambient temperature during the sampling period. The radon concentrations (in pCi/l) reported in Table 2 therefore represent the average ambient radon concentration for the sampling period for each specific sampling site location.

Radon Measurement Results

Ambient radon-222 concentrations measured in the various indoor working areas ranged from a low of 0.17 pCi/l in the Pelletizer Building to a high of 11 pCi/l in the Control Room of the Condenser and Fluid Bed Building. Harley (1975) estimated the average background concentration of radon in surface air of about 0.10 pCi/l in the northern hemisphere. Other authors (Pearson, 1967; United Nations, 1972) have reported average radon concentrations in the general environment ranging from 0.03 to 0.4 pCi/l. Ambient outdoor radon concentrations measured in areas associated with inactive uranium mill tailings piles

indicate average levels of 0.34 pCi/1 (Monticello, Utah), 0.38 pCi/1 (Salt Lake City, Utah), 0.51 pCi/1 (Durango, Colorado), and 0.83 pCi/1 (Grand Junction, Colorado), (U.S. Public Health Service, 1969).

Indoors, the radon levels are typically from three to four times the outdoor levels due to the entrapment within the structure of radon emanation from soils beneath the structure and also from radon emanated from the building materials, supplemented by radon infiltration from outside air. In general, radon levels inside structures are usually 0.6 to 1.2 pCi/1 for ventilation rates below four air changes per hour (Johnson et al., 1973). As shown in Table 2, the ambient radon concentrations measured indoors in

TABLE 2. AMBIENT RADON-222 CONCENTRATIONS

Date Time On	Off	Location Description	Radon Concentration* (pCi/1)
5/5/75 0958	1200	Pelletizer Building	0.17 ± 0.036
5/5/75 1027	1330	Control Rm -Calciner	0.63 ± 0.07
5/6/75 0902	1245	Control Room - Condenser & Fluid Bed	11 ± 0.28
5/6/75 1257	1429	Control Room (lower)- Proportioning Bldg.	1.4 ± 0.11
5/7/75 1223	1400	Control Room #3-4 Furnace Building	0.28 ± 0.047
5/9/75	Grab Sample	Carbon Monoxide Gas Sample	3.9 ± 1.4

* Radon-222 concentration ± two-sigma counting error term. To convert to pCi/m^3, multiply the pCi/1 value in the table by 1000.

the control rooms of the Condenser and Fluid Bed, and the Proportioning Buildings are higher than would normally be expected from natural terrestrial background sources alone. It is recommended that additional sampling for radon or radon progeny (expressed as Working Levels) be performed to better define both the area background and the in-plant levels.

Radon Dosimetry

The report by Swift et al. (1976) discusses various aspects of radon dosimetry including the degree of equilibrium between radon and its daughters, the fraction of unattached daughter ions and certain assumptions regarding exposure modes to tissues within the lung. For dose estimates in this study, a dose conversion factor of 4 mrem per year, for continuous exposure to 1 pCi/m^3, was used. This factor represents radon-222:polonium-218:lead-214:bismuth-214 concentrations of 1.0:0.90:0.51:0.35 pCi/m^3, respectively. This is equivalent to an average daughter product equilibrium ratio of 50 percent and an average ventilation rate of one air change per hour, which is typical in living accommodations with adequate ventilation. This dose is delivered to the basal cell nuclei of segmental bronchi which are estimated to lie 60 µm below the surface where the activity is deposited.

A quality factor of 10 for alpha particles has been used in the dose calculations. Therefore, continuous exposure to an assumed average background radon concentration of 100 pCi/m^3 (0.10 pCi/l as estimated by Harley, 1975) would result in a dose equivalent of 400 mrem per year. For a worker exposed for an entire work year (40 h/wk x 50 wk/year) to the maximum ambient measured radon concentration of 11,000 pCi/m^3, the dose equivalent would be about 10 rem per year.*

* 4 mrem/year/(pCi/m^3) x $\frac{2000 \text{ h Work Year}}{8760 \text{ h Continuous Exposure}}$ x 11,000 pCi/m^3 x 10^{-3} rem/mrem = 10 rem/year

Plant Discharge of Radon

One grab sample of the carbon monoxide gas, which is recycled through the system (Figure 1) but is ultimately discharged to the atmosphere, was obtained. The measured radon concentration in this sample was 3.9 pCi/l (Table 2).

THERMAL PROCESS SAMPLES

Appendix A discusses the radiochemical analytical techniques used for this study. Since the completion of the analyses of the following samples, a question has arisen as to the validity of the lead-210 radiochemical procedure. Although the procedure as outlined in Appendix A for the lead-210 analysis was fully tested and proved satisfactory for water sample analysis, there now appears to be some difficulty in obtaining consistent lead-210 recovery rates for solid samples such as the ore, briquettes, slag, and air filter samples. The inaccuracies of the lead-210 results relate to analysis of samples without the benefit of a tracer or carrier yield. Thus the assumed yield, which was assumed to be the same as that for test samples (i.e., 85 percent) was in error by up to a factor of five. Comparison of the lead-210 results as reported in Appendix C for the two EPA laboratories indicates that the EMSL lead analyses average about a factor of five lower than the actual lead-210 content (see Appendix E for data which limits the uncertainty for individual results). This error has also been noted in recent cross-check air sample results. Since the majority of samples have been consumed to obtain the results reported here, it is impossible to re-analyze these samples to resolve this question. Therefore, the lead-210 results which have been obtained using the EMSL procedure (Appendix A) have been reported here, but for solid samples, these results probably underestimate the real sample lead-210 activity by up to about a factor of five. Future lead-210 analyses will be conducted with a revised technique.

The problem of underestimating the lead-210 content is also reflected in the estimate of a sample's polonium-210 content, as discussed in Appendix A. The duplicate results reported for calcined nodules and slag (Appendix C, Table C-2 and C-3) indicate that, for instance, where the EMSL lead-210 values were used, the polonium-210 values may overestimate the actual polonium-210 content by up to a factor of ten or more (ranges from 1 up to about 20, but could be higher) i.e., if the reported EMSL lead-210 results are actually a factor of five too low. This error does not apply to the ore, calcined nodules, and slag samples; or the specific stack samples where the polonium was over an order of magnitude greater than the lead. Recognizing these difficulties encountered in obtaining valid lead-210 and polonium-210 results, the following sections describe the radiological evaluations of the Thermal Process.

The two sigma (95 percent) statistical counting error is reported for all of the radiochemistry data presented in this report. The techniques for calculating the counting errors are illustrated in Johns (1975). The error terms for data averages, the decay and ingrowth corrected polonium-210 results, and for the net air sample results represent error terms obtained using techniques for propagating error estimates as given in Appendix A and by Eadie and Bernhardt (1976).

It should be recognized that the statistical counting error only includes the uncertainty from the number of counts measured from the sample and instrument background. It is not an estimate of the total analytical uncertainty nor uncertainties in sample collection, preparation and/or aliquoting procedures. Evaluation of soil sampling and analytical techniques for plutonium by Bernhardt (1976) indicate that sampling and analytical variations of up to 50 percent and more (95 percent confidence level) are observable in many groups of data, especially at levels within about an order of magnitude of the analytical detection level. Results of aliquots for ore, nodule, and slag samples (Appendix C) indicate a general two-sigma aliquoting and analytical uncertainty from less than 10 percent to over 50 percent. Thus,

it should be recognized that the uncertainty of the results is greater than the counting error.

Table 3 presents the results of analyses of samples considered to be typical of the input to the thermal process. (Additional analyses for these samples are given in Appendix C.) The average radium-226 content of the Idaho phosphate rock (ore) was 26 pCi/g, and the calcined ore contained 25 pCi/g (Table 3). Radium-226 analysis of Florida ore average about 60 pCi/g (Guimond and Windham, 1975). Samples of silica and the two coke samples used in the Idaho plant had radium-226 contents of 1.7, 0.70, and 0.78 pCi/g, whereas samples from the Florida plants were 0.36 and 0.28 pCi/g, respectively. The Idaho silica and coke samples are at typical background concentrations of the naturally-occurring radionuclides (NCRP, 1975; Eisenbud, 1963), and are at least an order of magnitude lower in radioactivity content than the input ore.

Results of analyses of the products and by-products of the thermal process are shown in Table 4. The slag waste product samples had an average radium-226 content of 32 pCi/g (Appendix C, Table C-3) but had appreciably less polonium-210 and lead-210 as compared to the input ore (Table 3). (Specific gravity determinations on samples of Idaho phosphate ore and slag indicated average values of 2.7 and 2.8 g/cm^3, respectively.) Florida phosphate slag had an average radium-226 content of 56 pCi/g (Guimond and Windham, 1975).

TABLE 3. THE THERMAL PROCESS - INPUT TO ELECTRIC FURNACE
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/g

Radionuclide	Phosphate Rock**	Calcined Rock +	Silica	Coke	Coke Supplement
Ra-226	26 ± 19	25 ± 8.6	1.7 ± 0.24	0.78 ± 0.17	0.70 ± 0.16
Po-210+	22 ± 3.0	<2.65	2.6 ± 0.90	<1.3	0.98 ± 0.11
U-234	22 ± 2.0	23 ± 2.2	<0.86	<0.44	<0.95
U-235*	0.95 ± 0.40	1.0 ± 0.24	(0.07)	(<0.02)	(<0.03)
U-238	22 ± 3.2	24 ± 1.8	1.5 ± 1.4	<0.51	<0.59
Th-230	22 ± 4.1	23 ± 2.0	1.6 ± 0.53	<0.23	<0.20
Th-232	0.43 ± 0.12	0.47 ± 0.10	0.69 ± 0.37	<0.15	<0.22
Pb-210+	27 ± 12	18 ± 8.6	0.67 ± 0.55	2.4 ± 0.62	0.61 ± 0.54
Ra-228	0.89 ± 0.28	0.97 ± 0.48	<0.89	<0.95	<0.91

* U-235 values reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).

** Average results from both EMSL and EERF, except for lead-210 average of EERF values only; taken from Appendix C, Table C-1.

\+ Average results from both EMSL and EERF except for lead-210 and polonium-210 (corrected) which are average of EERF values only; taken from Appendix C, Table C-2.

TABLE 4. THE THERMAL PROCESS-PRODUCTS AND BY-PRODUCTS
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/g

Radionuclide	Phosphorus Product (Solid)	Ferrophosphorus	Fluid Bed Prills	Slag**
Ra-226	0.021 ± 0.020	0.27 ± 0.11	13 ± 0.65	32 ± 13
Po-210	-	<0.57	440 ± 27	<16
U-234	-	21 ± 5.2	<91	25 ± 6.7
U-235*	-	<0.65	(3.3)	1.1 ± 0.58
U-238	-	19 ± 4.9	<71	25 ± 7.0
Th-230	-	0.42 ± 0.27	<24	26 ± 11
Th-232	-	0.26 ± 0.20	<17	0.59 ± 0.29
Pb-210	0.21 ± 0.17	1.1 ± 0.58	52 ± 1.8	11 ± 7.9
Ra-228	<0.25	0.99 ± 0.94	1.9 ± 1.0	0.96 ± 0.46

*U-235 results reported in parenthesis are calculated based on the U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466). Blanks indicate no data.

**Average result of sample analyses completed at EMSL except for lead-210 and polonium-210 (corrected) are average of EERF values only; taken from Appendix C, Table C-3.

ENVIRONMENTAL SAMPLES

Grab water samples were obtained from the Portneuf River at locations both upstream and downstream from the liquid waste effluent discharge from both the Thermal Process and the Wet Process Plants. Prior to analysis, the suspended solids were allowed to "settle-out" and hence, both a liquid (soluble component) and a suspended (insoluble component) fraction of each water sample was processed. The water samples collected during this study are non-potable, and human ingestion of such water is improbable.

The radiochemical results for the Portneuf River samples are shown in Table 5. For the liquid fraction, the results of the upstream and downstream samples are essentially identical since they are within the two-sigma counting error term for each respective analysis. Similar analytical agreement was not obtained from the results of the suspended fraction analyses. Although the measured radium-226 content was lower in the downstream sample, the uranium and thorium contents were about an order of magnitude greater in the downstream versus the upstream sample. The higher radioactivity content in the suspended fraction may indicate the insolubility of waste-product effluents. It is emphasized that these results are based on analysis of only one grab sample from both the upstream and the downstream flow. Also, the measured radioactivity concentrations of both the liquid and suspended fractions of this river water sample are well within the typical background range for naturally-occurring radioactivity (Holtzman, 1964). Any future efforts to resolve whether there is any contribution of radioactivity to the stream should include sediment sampling.

Table 6 presents the results of a grab sample of rainwater runoff from the slag pile. As with the river water sample, both a liquid and a suspended fraction of the rainwater runoff sample

TABLE 5. PORTNEUF RIVER SAMPLES*
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM. LIQUID FRACTION IN UNITS OF pCi/l, AND SUSPENDED FRACTION IN UNITS OF pCi/g

Radionuclide	Upstream, East Bank - Liquid	Downstream, East Bank - Liquid	Upstream, East Bank-Suspended	Downstream, East Bank-Suspended
Ra-226	0.70 ± 0.35	0.54 ± 0.33	3.6 ± 1.7	1.8 ± 1.3
Po-210	ND	ND	-	<2.4
U-234	1.2 ± 0.22	1.2 ± 0.20	0.067 ± 0.019	0.70 ± 0.20
U-235	<0.031	<0.029	<0.0095	<0.10
U-238	0.55 ± 0.15	0.51 ± 0.11	0.067 ± 0.019	0.70 ± 0.20
Th-230	<0.062	<0.029	0.17 ± 0.048	1.6 ± 0.40
Th-232	<0.031	<0.029	0.17 ± 0.048	1.7 ± 0.40
Pb-210	2.7 ± 1.3	<1.4	-	6.1 ± 1.6
Ra-228	<3.6	<4.0	<20	-

*Grab sampled 5/8/75.
ND indicates non-detectable; blanks indicate no data.

TABLE 6. SLAG PILE RAINWATER RUNOFF*
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM

Radionuclide	Rainwater Runoff Liquid (pCi/l)	Rainwater Runoff Suspended (pCi/g)
Ra-226	0.70 ± 0.36	14 ± 0.72
Po-210	0.83 ± 0.75	ND
U-234	1.9 ± 0.25	4.0 ± 1.2
U-235	0.062 ± 0.031	0.14 ± 0.12
U-238	1.8 ± 0.25	4.1 ± 1.2
Th-230	0.12 ± 0.062	10 ± 0.25
Th-232	<0.031	0.49 ± 0.05
Pb-210	1.2 ± 0.36	150 ± 2.1
Ra-228	<3.6	0.98 ± 0.87

*Grab sampled 5/7/75. ND indicates non-detectable.

TABLE 7. NO. 4 FURNACE COOLING WATER (Sampled 5/8/75)
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/l

Radionuclide	Incoming Cooling Water	Outgoing-Bird Bath	Discharge Water-IWW
Ra-226	1.8 ± 0.52	0.30 ± 0.29	0.87 ± 0.35
Po-210	2.5 ± 1.3	<0.11	<0.82
U-234	1.8 ± 0.30	1.1 ± 0.20	2.6 ± 0.36
U-235	<0.05	<0.033	<0.04
U-238	0.70 ± 0.20	0.40 ± 0.10	0.96 ± 0.20
Th-230	0.10 ± 0.05	0.067 ± 0.033	<0.04
Th-232	<0.05	<0.033	<0.04
Pb-210	2.1 ± 0.57	1.5 ± 0.38	1.4 ± 0.48
Ra-228	<3.5	<3.4	<3.4

were analyzed, with the liquid fraction having background concentrations of radioactivity. However, the suspended fraction had elevated levels of radioactivity, perhaps also reflecting the water insolubility of these radionuclides in the slag waste material. That elevated levels of radioactivity were observed in the suspended fraction of the rainwater runoff does indicate that natural weathering factors, such as a normal rainstorm, may create a pathway for increasing the contamination of the local environs.

EFFLUENT SAMPLES

Cooling water samples from the No. 4 Furnace are shown in Table 7. All results are within the normal fluctuation of background concentrations of the naturally-occurring radionuclides in water.

Radiological analyses of water samples obtained from selected settling and evaporation ponds are shown in Table 8. These results are within normal fluctuations of background concentrations as measured in the Portneuf River samples (Table 5).

Results of analyses of effluents from various facility scrubber units are presented in Tables 9 and 10. Prior to analysis, solids in the sample were allowed to settle-out and hence, a liquid and a suspended fraction were analyzed. In every case, the suspended fraction contained appreciably higher quantities of radioactivity than did the liquid fraction, indicating the insolubility of these components. The highest radioactivity concentrations were measured in the samples from the Calciner Scrubber effluents (Table 9). Medusa scrubber samples also have slightly elevated radioactivity content compared to normal background levels (as reported for the Portneuf River samples, Table 5). Table 11 presents the results for a few miscellaneous samples.

TABLE 8. POND WATER - SETTLING AND EVAPORATION*

RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/l

Radionuclide	3-S Settling Pond	4-S Settling Pond	7-S Settling Pond	3-E Evaporation Pond	5-E Evaporation Pond**	6-E Evaporation Pond
Ra-226	0.90 ± 0.37	0.33 ± 0.30	1.3 ± 0.46	2.6 ± 0.62	0.60 ± 0.23	<0.29
Po-210	ND	66 ± 16	ND	160 ± 32	170 ± 47	32 ± 10
U-234	3.4 ± 1.8	0.68 ± 0.16	0.65 ± 0.20	1.2 ± 0.28	0.90 ± 0.25	0.68 ± 0.20
U-235	<0.62	<0.04	<0.05	<0.04	<0.05	<0.04
U-238	2.2 ± 1.2	0.40 ± 0.12	0.45 ± 0.15	0.80 ± 0.20	0.50 ± 0.15	0.48 ± 0.16
Th-230	1.0 ± 0.28	0.12 ± 0.04	0.25 ± 0.10	0.28 ± 0.16	<0.05	<0.04
Th-232	0.12 ± 0.08	<0.04	<0.05	<0.04	<0.05	<0.04
Pb-210	-	110 ± 3.7	100 ± 5.8	140 ± 3.4	100 ± 14	49 ± 2.3
Ra-228	<3.8	<3.5	<3.5	<3.4	<1.6	<3.4

* Settling ponds grab sampled 5/7/75; evaporation ponds grab sampled 5/8/75.
ND indicates non-detectable; blanks indicate no data.

** Grab sampled 8/13/75.

TABLE 9. CALCINER SCRUBBER EFFLUENTS*
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM. LIQUID FRACTION IN pCi/l, AND SUSPENDED FRACTION IN pCi/g

Radionuclide	Calciner Scrubber Pond Discharge (Liquid)	Calciner Scrubber Pond Discharge (Suspended)	No. 2 Calciner Scrubber Overflow (Liquid)	No. 2 Calciner Scrubber Overflow (Suspended)
Ra-226	7.8 ± 1.0	31 ± 1.2	4.6 ± 0.81	13 ± 0.93
Po-210	ND	-	21 ± 2.2	-
U-234	26 ± 1.6	-	13 ± 0.95	4.9 ± 0.40
U-235	1.0 ± 0.14	-	0.43 ± 0.10	0.18 ± 0.044
U-238	25 ± 1.6	-	12 ± 0.88	5.0 ± 0.41
Th-230	21 ± 0.89	-	0.23 ± 0.10	14 ± 0.56
Th-232	0.17 ± 0.086	-	<0.025	0.28 ± 0.078
Pb-210	13 ± 1.2	-	2.4 ± 0.34	-
Ra-228	<3.1	1.6 ± 1.1	<3.5	<1.8

*Grab sampled on 5/7/75.
ND indicates non-detectable; blanks indicate no data.

TABLE 10. SCRUBBER EFFLUENTS (Sampled 5/7/75)
RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.* LIQUID FRACTION IN pCi/l, and SUSPENDED FRACTION IN pCi/g

Radionuclide	Fluid Bed-Scrubber Liquor (Liquid)	Fluid Bed-Scrubber Liquor (Suspended)	Medusa Scrubber Liquor-No. 4-E, tapping (Liquid)	Medusa Scrubber Liquor-No. 4-E, tapping (Suspended)	Medusa Scrubber Liquor-No. 4-W, not tapping (Liquid)	Medusa Scrubber Liquor-No. 4-W, not tapping (Suspended)
Ra-226	2.3 ± 0.56	8.3 ± 0.53	2.5 ± 0.59	9.2 ± 7.7	1.1 ± 0.44	24 ± 10
Po-210	ND	ND	ND	-	11 ± 1.4	35 ± 7.4
U-234	0.72 ± 0.20	0.87 ± 0.32	2.2 ± 0.23	-	1.1 ± 0.14	0.60 ± 0.20
U-235	<0.04	<0.11	0.057 ± 0.029	-	<0.029	<0.10
U-238	0.44 ± 0.12	0.83 ± 0.35	1.4 ± 0.17	-	0.51 ± 0.086	0.50 ± 0.20
Th-230	<0.04	2.4 ± 0.18	0.54 ± 0.11	-	0.23 ± 0.086	0.60 ± 0.30
Th-232	<0.04	0.11 ± 0.04	<0.029	-	<0.029	<0.10
Pb-210	46 ± 1.9	1900 ± 19	210 ± 3.2	-	1.6 ± 0.34	31 ± 3.3
Ra-228	<3.7	<0.88	6.5 ± 3.8	<67	4.1 ± 3.5	<170

*ND indicates non-detectable; blanks indicate no data.

TABLE 11. MISCELLANEOUS SAMPLES (Sampled 5/8/75)

RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/l

Radionuclide	Fluid Bed Recycle - Furnace Slurry (Liquid)	10,000 Gallon Sump - PHOS Water from Furnace (Liquid)
Ra-226	9.8 ± 0.82	1.3 ± 0.30
Ra-228	2.1 ± 1.1	<1.8
Pb-210	<2.1	43 ± 2.5

AIRBORNE PARTICULATE RADIOACTIVITY

Background Airborne Radioactivity Concentrations

Background airborne radioactivity concentrations were not specifically obtained for the Pocatello area. However, as will be discussed shortly, the airborne concentrations measured in the Technical Building Library may be considered as representative of background conditions for the purposes of this report. Other authors have reported background airborne levels. Poet et al. (1972) reported the average polonium-210 and lead-210 concentrations in ground level air to be 0.001 and 0.01 pCi/m^3, respectively. NCRP (November, 1975) summarizes reports by geographic areas of the normal polonium-210 and lead-210 air concentrations. In Colorado, the reported polonium-210 concentrations ranged from 0.00005 to 0.003 pCi/m^3, whereas the lead-210 ranged from 0.001 to 0.021 pCi/m^3. The highest reported lead-210 concentrations were 0.026 pCi/m^3 for several locations in Illinois and Utah.

NCRP (November, 1975) also discusses the background level of airborne concentrations of the other naturally-occurring radionuclides. Natural uranium (uranium-234 and -238) concentrations ranged from 120 aCi/m^3 (one attocurie is 10^{-18} curie or 10^{-6} pCi) near Chicago, Illinois, to a reported high of about 400 aCi/m^3 for several sites in New York State. Typical radium-226 concentrations are usually less than 100 aCi/m^3. Sedlet et al. (1973) reported thorium-232 and thorium-230 concentrations of 30 and 45 aCi/m^3, respectively, near Chicago, Illinois.

In-Plant Air Sampler

Air samples were obtained in the in-plant areas using a portable air sampling unit.* This unit has a carbon vane vacuum pump with a regulator which permits constant air flow sampling. Usually, sample collections were made at two cubic feet per

* Regulated Air Sampler, Model RAS-1, Eberline Instrument Corporation, Santa Fe, New Mexico

minute (CFM) using a 47-millimeter diameter, Type E glass fiber filter. This corresponds to a linear flow rate of about 106 feet per minute. The air sample dust load was determined by measuring the mass of material collected on each filter.

Airborne particulate sampling was conducted in several working areas within the Thermal Process Plant using the portable air sampler. The radioactivity concentrations of the naturally-occurring radionuclides are usually expressed as picocuries per cubic meter of sampled air (pCi/m^3). Dust loading of the air filter samples was also determined and the specific activity of the dust, expressed as pCi/g, is also given. The solubility of airborne particulate matter was not determined in this study.

Gross Versus Net Results

Eadie and Bernhardt (1976) have reported that radiochemical analyses of blank glass fiber filters (4-inch diameter) indicate appreciable quantities of naturally-occurring radioactivity. Comparison of these results with those of other facilities indicates that the reported activity is not only due to the composition of the filter media itself, but also due to contaminants in reagents, glassware, the general background level of contaminants, etc.

Table 12 presents data on the radioactivity content of blank glass fiber filters (4-inch diameter) and the extrapolated activity content for the 47-mm diameter filters which were used in this study. In order to account for this radioactivity content associated with blank filter analyses, the appropriate blank filter activity (Table 12) has been subtracted from the measured gross analytical result to obtain a "net" result. No blank subtractions have been made for the three radionuclides (radium-228, polonium-210, and lead-210) which are at the analytical minimum detectable activity (MDA) levels. Gross analytical results for all of the in-plant air samples are shown in Appendix B.

TABLE 12. RADIOACTIVITY CONTENT OF BLANK GLASS FIBER FILTERS (pCi per filter)+

Radionuclide	4-inch Diameter*	47-mm Diameter**
Ra-226	0.35 ± 0.09	0.07 ±0.019
Po-210	<0.17	<0.036
U-234	0.10 ± 0.03	0.021 ±0.0064
U-235***	(0.0035 ± 0.0010)	(0.00075 ±0.00021)
U-238	0.08 ± 0.02	0.017 ±0.0043
Th-230	0.20 ± 0.08	0.043 ±0.017
Th-232	0.13 ± 0.02	0.028 ±0.0043
Pb-210	<0.32	<0.068
Ra-228	<1.6	<0.34

+ Average pCi per filter with standard error term about this mean based on the t-distribution at the 95 percent confidence level.
* Taken from Eadie and Bernhardt, 1976.
** Extrapolated value based on the area ratio between 47-mm and 4-inch diameter filters of 0.214 times the 4-inch diameter activity content. Average 47-mm filter mass ± two standard deviations was 0.1249 ± 0.0013 grams.
***Calculated U-235 content based on U-235 to U-238 natural activity ratio of 1:21.45 (0.0466).

In-Plant Air Sampling Net Results

Tables 13 to 17 present the net radionuclide concentrations obtained for the various in-plant sampling locations. As previously noted, the lead-210 results are underestimated by up to a factor of five, which results in a potential overestimate of about a factor of ten or more for the polonium-210 results. Data which narrows the uncertainty for specific samples are given in Appendix E.

The highest airborne radioactivity was measured in the area of the south side discharge of the Calciner (Table 13). There, the measured airborne concentrations are orders of magnitude greater than results obtained in the Technical Building Library (Table 17). For example, because of the higher airborne dust levels, the radium-226 and total uranium airborne concentrations (i.e., pCi/m^3) measured in the Calciner were 3800 and 1600 times the respective levels measured in the Technical Building Library. Non-detectable levels of polonium-210 were measured in the Technical Building Library (Table 17) compared to the highest value of 66 pCi/m^3 measured in the area of the Calciner (Table 13).

TABLE 13. AIRBORNE PARTICULATE SAMPLING IN THE CALCINER AREA
NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM

Radionuclide	South Side Discharge,* Calciner Belt #1 Dump Area - 1218 to 1500 hrs. (5/5/75)		No. 1 Control Room** 1027 to 1520 hrs. (5/5/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	2.7 ± 0.18	38 ± 2.4	0.015 ± 0.010	19 ± 13
Po-210+	66 ± 4.9	560 ± 70	<0.0078	<88
U-234	1.5 ± 0.12	20 ± 1.7	0.0079 ± 0.0023	10 ± 3.1
U-235***	0.044 ± 0.014	0.62 ± 0.20	(<0.00028)	(<0.37)
U-238	1.5 ± 0.12	21 ± 1.7	0.0058 ± 0.0023	7.9 ± 3.1
Th-230	2.8 ± 0.15	39 ± 2.1	0.0067 ± 0.0046	8.7 ± 6.2
Th-232	0.051 ± 0.022	0.72 ± 0.30	0.0029 ± 0.0023	4.0 ± 3.1
Pb-210+	0.40 ± 0.28	120 ± 9.5	0.17 ± 0.066	210 ± 83
Ra-228+	<0.23	<3.2	<0.085	<120

+Gross analytical results.
*Sampled volume of $7.9m^3$, with a dust particulate load of 0.5663g.
**Sampled volume of $17.6m^3$, with a dust particulate load of 0.0131g.
***U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).

TABLE 14. AIRBORNE PARTICULATE SAMPLING IN THE PROPORTIONING BUILDING
NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM

Radionuclide	Lower Level, 1st Pillar* West of Shack, 0947 to 1309 hrs. (5/6/75)		Lower Level Control Room** 0944 to 1316 hrs. (5/6/75)		Upper Level, Top West+ 1140 to 1315 hrs. (5/6/75)		Upper Level Control Room++ 0930 to 1411 hrs. (5/6/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.37 ± 0.045	15 ± 1.8	0.12 ± 0.027	16 ± 3.5	2.9 ± 0.21	16 ± 1.2	0.48 ± 0.054	30 ± 3.4
Po-210°	5.3 ± 0.89	210 ± 36	<2.7	350 ± 71	<3.4	<21	3.8 ± 0.94	240 ± 60
U-234	0.30 ± 0.036	12 ± 1.5	0.095 ± 0.017	13 ± 2.2	2.8 ± 0.23	16 ± 1.3	0.29 ± 0.034	18 ± 2.1
U-235	0.014 ± 0.0069	0.54 ± 0.27	0.0044 ± 0.0039	0.58 ± 0.51	0.13 ± 0.027	0.71 ± 0.15	0.012 ± 0.0058	0.73 ± 0.35
U-238	0.32 ± 0.038	13 ± 1.5	0.11 ± 0.019	15 ± 2.4	2.8 ± 0.23	16 ± 1.3	0.32 ± 0.035	19 ± 2.2
Th-230	0.72 ± 0.062	29 ± 2.5	0.21 ± 0.034	27 ± 4.3	4.9 ± 0.25	27 ± 1.4	0.50 ± 0.048	31 ± 3.0
Th-232	0.035 ± 0.014	1.4 ± 0.58	<0.00056	<0.064	0.11 ± 0.036	0.62 ± 0.20	0.013 ± 0.0083	0.79 ± 0.51
Pb-210°	0.63 ± 0.11	25 ± 4.5	0.24 ± 0.097	30 ± 12	6.6 ± 0.48	37 ± 2.7	0.74 ± 0.10	45 ± 6.5
Ra-228°	<0.15	<6.0	<0.12	<15	<0.39	<2.2	<0.11	<7.0

*Sampled volume of $12.4m^3$, with a dust particulate load of 0.3109g.
**Sampled volume of $12.5m^3$, with a dust particulate load of 0.0978g.
+Sampled volume of $4.44m^3$, with a dust particulate load of 0.7853g.
++Sampled volume of $14.7m^3$, with a dust particulate load of 0.2387g.
°Gross analytical result.

TABLE 15. AIRBORNE PARTICULATE SAMPLING IN THE PELLETIZER BUILDING
NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM

Radionuclide	Operator Level, Middle** Lower Level, 1000 to 1537 hrs.(5/5/75)		Upper Level, Top East+ 1003 to 1530 hrs.(5/5/75)		Upper Control Room++ 1006 to 1528 hrs.(5/5/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.28 ± 0.033	20 ± 2.3	0.40 ± 0.044	24 ± 2.5	0.17 ± 0.03	28 ± 4.9
Po-210*	1.1 ± 0.046	82 ± 31	1.6 ± 0.042	86 ± 24	ND	ND
U-234	0.45 ± 0.042	32 ± 2.9	0.26 ± 0.029	15 ± 1.7	0.099 ± 0.015	17 ± 2.6
U-235	0.023 ± 0.0096	1.6 ± 0.68	0.014 ± 0.0058	0.80± 0.33	0.0052± 0.0034	0.85± 0.57
U-238	0.24 ± 0.03	17 ± 2.1	0.25 ± 0.028	14 ± 1.6	0.099 ± 0.016	17 ± 2.6
Th-230	0.50 ± 0.044	35 ± 3.1	0.53 ± 0.05	30 ± 2.9	0.19 ± 0.028	32 ± 4.7
Th-232	0.019 ± 0.0089	1.2 ± 0.63	0.023 ± 0.011	1.4 ± 0.66	0.012 ± 0.0079	1.9 ± 1.3
Pb-210*	0.66 ± 0.10	46 ± 7.1	0.60 ± 0.092	37 ± 5.7	0.30 ± 0.07	50 ± 12
Ra-228*	<0.11	<7.7	<0.10	<6.0	<0.095	<16

*Gross analytical results. ND indicates non-detectable.
**Sampled volume of 18.7m^3, with a dust particulate load of 0.2666g.
+Sampled volume of 14.3m^3, with a dust particulate load of 0.2502g.
++Sampled volume of 18.0m^3, with a dust particulate load of 0.1083g.

TABLE 16. AIRBORNE PARTICULATE SAMPLING IN THE FURNACE BUILDING
NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM

Radionuclide	Control Room Level * Between Furnace 3 & 4 0906 to 1606 hrs.(5/7/75)		Tapping Level** Between Furnace 3 & 4 0910 to 1607 hrs.(5/7/75)		Control Room for + Furnace 3 & 4, Behind Panel 0853 to 0851 hrs.(5/7 to 8/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.044 ± 0.012	16 ± 4.4	0.011 ± 0.0063	6.8 ± 4.0	0.0082 ± 0.0030	11 ± 4.0
Po-210++	0.63 ± 0.23	229 ± 84	<0.048	<69	0.16 ± 0.057	210 ± 79
U-234	0.010 ± 0.0041	3.5 ± 1.5	0.0086 ± 0.0046	5.6 ± 3.0	0.0022 ± 0.0012	2.9 ± 1.6
U-235 +++	<0.00094	<0.34	(<0.00048)	(<0.32)	(<0.00019)	(<0.25)
U-238	0.010 ± 0.0041	3.5 ± 1.5	0.010 ± 0.0052	6.7 ± 3.4	0.0040 ± 0.0016	5.2 ± 2.1
Th-230	0.018 ± 0.0079	6.4 ± 2.8	0.0071 ± 0.0053	4.2 ± 3.4	0.0031 ± 0.0017	4.1 ± 2.6
Th-232	0.0054 ± 0.0047	2.0 ± 1.7	<0.0030	<1.9	<0.00056	<0.75
Pb-210++	0.27 ± 0.054	98 ± 20	0.25 ± 0.055	150 ± 34	0.027 ± 0.013	36 ± 18
Ra-228++	<0.069	<25	<0.067	<44	<0.019	<25

*Sampled volume of 24.3m^3, with a dust particulate load of 0.0678g.
**Sampled volume of 23.4m^3, with a dust particulate load of 0.0370g.
+Sampled volume of 84m^3, with a dust particulate load of 0.0623g.
++Gross analytical result
+++U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).

TABLE 17. AIRBORNE PARTICULATE SAMPLING
NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM

Radionuclide	Technical Building* Library- 0935 (5/5/75) to 1421 hrs.(5/6/75)		Condenser and Fluid Bed** Control Room- 0904 to 1444 hrs.(5/6/75)		Old Kiln Building (Maintenance)+ Open Area - East End 0912 to 1412 hrs.(5/6/75)		P-4 Loading Area++ 0933 to 1438 hrs.(5/8/75)
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3
Ra-226	<0.00072	<12	0.0084 ± 0.0081	6.7 ± 6.4	0.27 ± 0.034	24 ± 3.0	0.0024 ± 0.0008
Po-210°	ND	ND	0.29 ± 0.25	250 ± 210	43 ± 4.9	3900 ± 430	0.038 ± 0.027
U-234	0.00077 ± 0.00063	12 ± 9.9	0.014 ± 0.0038	12 ± 3.1	0.20 ± 0.025	18 ± 2.1	0.0019 ± 0.00058
U-235°°	(<0.000050)	(<0.84)	(<0.00047)	(<0.41)	0.0079± 0.0044	0.69 ± 0.38	(<0.00010)
U-238	0.0011 ± 0.00063	18 ± 9.9	0.010 ± 0.0038	8.7 ± 3.1	0.21 ± 0.025	18 ± 2.2	0.0025 ± 0.00065
Th-230	0.0016 ± 0.00065	25 ± 10	0.015 ± 0.0057	12 ± 4.7	0.33 ± 0.037	29 ± 3.3	0.0037 ± 0.00022
Th-232	<0.000095	<1.4	0.0044 ± 0.0038	3.6 ± 3.1	0.022 ± 0.011	1.9 ± 0.95	0.000068± 0.00022
Pb-210°	0.019 ± 0.0086	320 ± 140	0.20 ± 0.059	160 ± 47	0.58 ± 0.089	51 ± 7.9	0.018 ± 0.0038
Ra-228°	<0.011	<190	<0.081	<66	<0.092	<8.1	<0.0045

* Sampled volume of 130m^3, with a dust particulate load of 0.0081g.

** Sampled volume of 21m^3, with a dust particulate load of 0.0256g.

\+ Sampled volume of 16.4m^3, with a dust particulate load of 0.1862g.

++ Sampled volume of 346m^3. Sample obtained on 8 x 10-inch glass fiber filter using only the final stage of the high volume cascade impactor. Blank filter activity of 8 x 10-inch filter (Table 23) has been subtracted from gross analytical results (Eadie and Bernhardt, 1976).

° Gross analytical results. ND indicates non-detectable.

°° U-235 reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).

Although the library had the lowest airborne radioactivity measured in this study, even these levels were slightly elevated compared to reported background levels reported by other authors, but for the purposes of this study the results measured in the Technical Building Library (Table 17) shall be considered as background levels.

Specific Activity of Airborne Particulate Matter

The specific activity of airborne particulate matter (reported in units of pCi/g) was also determined for the portable air sampler filters. These results also show the disequilibrium between the individual radionuclides of the uranium-238 decay series, the most noticeable being the polonium-210 and lead-210 contents compared to the other decay chain members. (Thorium-232 and radium-228 are members of a different decay series.)

The highest polonium-210 activity was 3900 ± 430 pCi/g in the Old Kiln Building (Table 17) with several sampling locations having essentially a non-detectable polonium-210 level (Tables 15 and 17). The highest radium-226 activity of 38 ± 2.4 pCi/g was obtained in the Calciner (Table 13) and the lowest activity was 6.7 ± 6.4 pCi/g in the Condenser and Fluid Bed Control Room (Table 17).

These specific activity determinations of the airborne particulate matter may also be compared to the results of analyses of the input products to the thermal process (Table 3). The activity of 22 pCi/g in phosphate rock was the highest concentration measured in any of the input product samples for polonium-210, yet specific activity determinations of the airborne particulate samples were at least a factor of three greater than this - except for the 21 pCi/g polonium-210 measured in the Upper Level, Top West of the Proportioning Building. The specific activity concentrations for the other radionuclides in the input products are generally comparable to the airborne particulate sample results.

PARTICLE SIZE CHARACTERIZATION

Cascade Impactor Sampling

High volume cascade impactors* were used to measure the size distribution of airborne particulate matter for both the indoor and outdoor environments. The impactor filter stages attach to a high volume air sampler** which is electronically controlled*** to operate at a flow rate of 40 cubic feet per minute (CFM). This corresponds to a linear air velocity of about 72 feet per minute through the final 8 x 10-inch filter stage.

Whatman #41 paper filters were used for each impaction stage. Glass fiber (Type E) was used for the final 8 x 10-inch filter. Typical particle size ranges for each filter stage, as reported by the manufacturer, are:

Stage No.	Equivalent Aerodynamic Diameter at 50 Percent Collection Efficiency (micrometer)
1	Greater than 7.2 μm
2	3.0 - 7.2
3	1.5 - 3.0
4	0.95- 1.5
5	0.49- 0.95
Final Filter	Less than 0.49

"Equivalent aerodynamic diameter" is defined as the size of a spherical particle of unit density which has the same terminal settling velocity as the sampled particle. Radioactivity analysis

* Model 252 Series, six stage, cascade impactor from Tech Ecology, Inc. (Now produced and marketed by Sierra Instruments, Inc.)

** Model GMWL-2000, High Volume Air Sampling System from General Metal Works, Inc.

*** Model 310/310A, High Volume Constant Flow Controller from Sierra Instruments, Inc.

of each impactor stage provides an indication of the activity content for the various particle size ranges. The mass of dust particulate loading (in grams) for each filter stage was not determined for this study.

Gross Results of Impactor Sampling

The cascade impactor sampler was used at four locations in the Thermal Process Plant - the Calciner, Proportioning, Pelletizer, and Furnace Buildings. Gross analytical results for these samples are presented in Tables 18 to 21. Tables 22 and 23 show the results of radiochemical analyses of blank impactor filters - the slotted Whatman #41 paper and the 8 x 10-inch glass fiber filters, respectively. Since most of the blank slotted paper filters contained radioactivity concentrations at the minimum detectable activity (MDA) levels for the analysis, a blank filter subtraction has not been performed for all the radionuclides for the impactor samples. As noted in subsequent discussions, tabulations of net results have been used in some of the data plots. The previously noted reservations concerning the lead-210 and polonium-210 data also apply to these results (i.e., lead-210 may be low by a factor of 5 or more and polonium-210 high by a factor of 10). Data for narrowing the uncertainty on specific samples are given in Appendix E.

Table 24 provides a summary of the gross results of the cascade impactor sampling for the four sampling locations. The gross results for the Calciner (Table 18) indicate that the majority of activity (roughly 53 percent of the total arithmetic activity collected) was of the particle size fraction less than 0.5 μm equivalent aerodynamic diameter (i.e., the activity collected on the final filter stage). Gross results for the other locations also indicate that a large fraction of the activity is in the sub-micron particle size range.

Comparison of Air Sampling Results

The activity summation for all filters of the cascade impactor may be compared to the activity concentrations determined

TABLE 18. CASCADE IMPACTOR SAMPLING IN THE CALCINER AREA*
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/m^3.

Radionuclide	Filter 1	Filter 2	Filter 3	Filter 4	Filter 5	Final Filter	Activity Summation For All Filters
Ra-226	0.18 ± 0.02	0.079 ± 0.015	0.043 ± 0.01	0.042 ± 0.0097	0.051 ± 0.011	0.38 ± 0.028	0.78
Po-210	11 ± 1.1	3.0 ± 0.39	1.6 ± 0.26	1.1 ± 0.24	1.6 ± 0.25	21 ± 1.7	39
U-234	0.18 ± 0.018	0.041 ± 0.0074	0.022 ± 0.0053	0.056 ± 0.01	0.046 ± 0.0083	0.38 ± 0.03	0.73
U-235	0.011 ± 0.0035	<0.0011	<0.00077	<0.0026	<0.0016	0.0099 ± 0.0033	<0.027
U-238	0.17 ± 0.017	0.039 ± 0.0072	0.018 ± 0.0047	0.046 ± 0.01	0.019 ± 0.0054	0.38 ± 0.03	0.67
Th-230	0.32 ± 0.0013	0.07 ± 0.013	0.077 ± 0.013	0.082 ± 0.015	0.049 ± 0.011	0.70 ± 0.038	1.30
Th-232	0.0077 ± 0.0045	0.007 ± 0.0042	0.0037 ± 0.0028	<0.0026	<0.0032	0.019 ± 0.0064	<0.043
Pb-210	0.57 ± 0.057	0.28 ± 0.056	0.14 ± 0.041	0.13 ± 0.038	0.065 ± 0.036	1.1 ± 0.071	2.3
Ra-228	<0.058	<0.058	<0.058	<0.059	<0.06	<0.059	<0.35
Percentage of Total Activity	28%	8.0%	4.3%	3.4%	4.2%	53%	–

*Sample collected 5/5/75, 1347 to 1415 hours, total volume sampled of 31.3 m^3. Sampled at South Side Discharge, Calciner Belt #1 Dump Area.

TABLE 19. CASCADE IMPACTOR SAMPLING IN THE PROPORTIONING BUILDING*
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/m^3.

Radionuclide	Filter 1	Filter 2	Filter 3	Filter 4	Filter 5	Final Filter	Activity Summation For All Filters
Ra-226	0.24 ± 0.032	0.036 ± 0.012	0.041 ± 0.014	0.053 ± 0.016	0.054 ± 0.011	0.20 ± 0.03	0.62
Po-210	0.57 ± 0.36	<0.46	<0.11	<1.2	<0.27	0.65 ± 0.37	3.3
U-234	0.17 ± 0.02	0.055 ± 0.013	0.018 ± 0.0075	0.028 ± 0.0086	0.018 ± 0.0075	0.15 ± 0.02	0.44
U-235	0.005 ± 0.0025	0.0039 ± 0.0037	<0.0025	<0.0032	<0.0025	<0.0044	<0.022
U-238	0.15 ± 0.018	0.033 ± 0.0098	0.018 ± 0.0075	0.019 ± 0.0071	0.015 ± 0.0075	0.16 ± 0.021	0.40
Th-230	0.25 ± 0.028	0.087 ± 0.02	0.015 ± 0.0075	0.022 ± 0.014	0.023 ± 0.0075	0.28 ± 0.035	0.68
Th-232	0.01 ± 0.005	<0.0029	<0.0025	0.015 ± 0.011	<0.0025	0.011 ± 0.0078	<0.044
Pb-210	0.33 ± 0.082	0.18 ± 0.076	0.11 ± 0.073	0.14 ± 0.075	0.11 ± 0.074	0.49 ± 0.086	1.4
Ra-228	<0.12	<0.12	<0.12	<0.12	<0.12	<0.10	<0.70
Percentage Of Total Activity	24%	13%	5.8%	21%	8.1%	27%	

*Sample collected 5/6/75, 1347 to 1401 hours, total volume sampled of 15.9m^3. Sampled at lower level, 1st pillar west of shack.

TABLE 20. CASCADE IMPACTOR SAMPLING IN THE PELLETIZER BUILDING*
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR, IN pCi/m^3.

Radionuclide	Filter 1	Filter 2	Filter 3	Filter 4	Filter 5	Final Filter	Activity Summation For All Filters
Ra-226	0.013 ± 0.005	0.016 ± 0.005	0.013 ± 0.005	0.011 ± 0.0041	0.0088 ± 0.0039	0.034 ± 0.0074	0.095
Po-210	ND	0.0051 ± 0.00085	ND	<0.070	ND	0.14 ± 0.095	0.22
U-234	0.016 ± 0.0037	0.0098 ± 0.0032	0.0098 ± 0.003	0.0067 ± 0.0028	0.0089 ± 0.0035	0.045 ± 0.0069	0.096
U-235	<0.00091	<0.001	<0.00078	<0.0015	<0.0018	0.0026 ± 0.0016	<0.0086
U-238	0.019 ± 0.004	0.0076 ± 0.0026	<0.00055	0.0085 ± 0.0032	0.011 ± 0.0035	0.046 ± 0.007	0.093
Th-230	0.026 ± 0.0062	0.012 ± 0.0044	0.020 ± 0.0054	0.017 ± 0.0051	0.011 ± 0.0053	0.071 ± 0.010	0.16
Th-232	<0.0014	<0.0014	<0.0016	<0.002	<0.0018	0.0027 ± 0.00068	<0.011
Pb-210	0.073 ± 0.027	0.025 ± 0.017	0.058 ± 0.026	<0.018	0.049 ± 0.027	0.088 ± 0.026	0.31
Ra-228	<0.04	<0.04	<0.039	<0.04	<0.04	<0.041	<0.24
Percentage Of Total Activity	15%	9.6%	12%	14%	11%	38%	

*Sample collected 5/5/75, 1102 to 1141 hours, total volume sampled of 45.5 m^3. Sampled at Operator Level, middle of the Pelletizer Lower Level. ND indicates non-detectable.

TABLE 21. CASCADE IMPACTOR SAMPLING IN THE FURNACE BUILDING*
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, IN pCi/m^3.

Radionuclide	Filter 1	Filter 2	Filter 3	Filter 4	Filter 5	Final Filter	Activity Summation For All Filters
Ra-226	0.0025 ± 0.0014	0.0045 ± 0.0018	0.0019 ± 0.0014	0.0058 ± 0.002	0.0022 ± 0.0013	0.0073 ± 0.0023	0.024
Po-210	ND	ND	<0.0033	<0.031	<0.048	0.12 ± 0.055	<0.20
U-234	0.0029 ± 0.0011	0.0024 ± 0.0011	0.0013 ± 0.00073	0.0019 ± 0.00087	0.0013 ± 0.00072	0.0026 ± 0.00099	0.012
U-235*	(0.000065)	(0.000047)	(0.00003)	(0.000051)	(0.000084)	(0.00013)	(0.00041)
U-238	0.0014 ± 0.00081	<0.001	0.00064 ± 0.00055	0.0011 ± 0.00065	0.0018 ± 0.00082	0.0027 ± 0.001	<0.0086
Th-230	0.0037 ± 0.0015	0.0036 ± 0.0016	0.0026 ± 0.0015	0.0025 ± 0.0015	<0.001	0.0052 ± 0.002	<0.019
Th-232	0.00073 ± 0.00066	0.00096 ± 0.00087	<0.00041	<0.0011	<0.00084	<0.0014	<0.0054
Pb-210	0.03 ± 0.011	0.034 ± 0.011	0.027 ± 0.011	0.039 ± 0.012	0.064 ± 0.013	0.061 ± 0.013	0.26
Ra-228	<0.016	<0.016	<0.016	<0.016	<0.016	<0.016	<0.096
Percentage Of Total Activity	9.2%	10%	8.5%	16%	22%	35%	

*Sample collected 5/7/75, 1330 to 1505 hours, total volume sampled of 108m^3. Sampled at Control Room level, between furnace 3 and 4 above slag slough. U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).
ND indicates non-detectable.

TABLE 22. RADIOACTIVITY CONTENT OF BLANK IMPACTOR WHATMAN #41 PAPER FILTER (SLOTTED)
(pCi ± TWO-SIGMA COUNTING ERROR TERM PER FILTER)

Filter	Ra-226	Po-210	U-234	U-235*	U-238	Th-230	Th-232	Pb-210	Ra-228
1	<0.13	<5.1	<0.36	<0.014	<0.31	-	-	<1.3	<1.7
2	-	<3.1	<0.95	<0.023	<0.50	<0.068	<0.068	<1.2	<2.0
3	0.23 ± 0.17	12 ± 6.4	1.2 ±0.85	<0.023	<0.50	<0.070	<0.034	<1.2	<2.2
4	0.37 ± 0.17	7.8 ± 4.2	<0.67	<0.030	<0.64	<0.045	<0.034	<1.1	<2.0
5	0.17 ± 0.14	10 ± 5.0	0.70 ±0.62	<0.033	<0.70	<0.062	<0.048	<1.1	<2.0
Average of All Results**	<0.23	<7.6	<0.78	<0.025	<0.53	<0.061	<0.046	<1.2	<2.0

*U-235 calculated based on natural U-235 to U-238 activity ratio of 1 to 21.45 (0.0466). Blanks indicate no data.
**All average results are reported as less than values since most of the analytical results of the individual filter are less than values.

TABLE 23. RADIOACTIVITY CONTENT OF BLANK GLASS FIBER FILTERS (8 BY 10-INCH)
(pCi ± TWO-SIGMA COUNTING ERROR TERM PER FILTER)

Filter	Ra-226	Po-210	U-234	U-235*	U-238	Th-230	Th-232
1	0.21 ± 0.08	0.64 ± 0.38	0.29 ± 0.10	0.012 ± 0.0045	0.26 ± 0.096	0.33 ± 0.11	0.14 ± 0.072
2	0.15 ± 0.07	<0.53	0.34 ± 0.12	0.014 ± 0.0051	0.31 ± 0.11	0.33 ± 0.13	0.15 ± 0.08
3	0.57 ± 0.13	LOST	0.35 ± 0.16	0.018 ± 0.0084	0.38 ± 0.18	0.44 ± 0.18	0.16 ± 0.15
4	0.32 ± 0.10	<0.54	0.36 ± 0.13	0.017 ± 0.0061	0.36 ± 0.13	0.40 ± 0.14	0.18 ± 0.088
5	0.17 ± 0.07	<0.58	0.25 ± 0.11	0.0079 ± 0.0041	0.17 ± 0.088	0.32 ± 0.13	0.18 ± 0.096
6	0.23 ± 0.08	<0.58	0.30 ± 0.13	0.0075 ± 0.0061	0.16 ± 0.13	0.22 ± 0.11	0.13 ± 0.072
Average of All Results**	0.28 ± 0.16	<0.57	0.32 ± 0.04	0.013 ± 0.0044	0.27 ± 0.094	0.34 ± 0.076	0.16 ± 0.021

*U-235 calculated based on natural U-235 to U-238 activity ratio of 1 to 21.45 (0.0466).
**Average of all results with standard error about this mean based in the t-distribution at the 95 percent confidence level.

TABLE 24. SUMMARY OF GROSS RESULTS OF THE CASCADE IMPACTOR SAMPLING

Location	Percentage of Total Radioactivity per Particle Size Range					
	Greater than 7.2 μm	3.0 to 7.2 μm	1.5 to 3.0 μm	0.95 to 1.5 μm	0.49 to 0.95 μm	Less than 0.49 μm
Calciner	28	7.9	4.3	3.4	4.2	53
Proportioning Building	24	13	5.8	21	8.1	27
Pelletizer Building	15	9.6	12	14	11	38
Furnace Building	9.2	10	8.5	16	22	35
Arithmetic Average Percentage	19	10	7.7	14	11	38

from the portable air sampler for the same sampling location. Good agreement has been obtained between the results of the radioactivity determinations using these two different sampling systems, even though the samples were not necessarily obtained during simultaneous time periods.

For the Calciner, impactor results (Table 18) are roughly one-half of the activity as measured using the portable air sampler (first column of Table 13). In both cases, the results indicate that polonium-210, and to a lesser extent lead-210 activities are present in concentrations in excess of the normal equilibrium concentrations of the other decay chain members. Of all the sampling locations, the highest airborne radioactivity measurements were obtained in the Calciner, regardless of the air sampling system used (this conclusion recognizes the previously denoted uncertainties for the lead-210 and polonium-210 results).

In the Proportioning Building, the cascade impactor results (Table 19) are within a factor of two of the individual radionuclide concentrations measured using the portable air sampler (first column of Table 14). Except for the elevated polonium-210 activity, the decay chain products appear to be in equilibrium. Cascade impactor results in the Pelletizer Building (Table 20) are roughly one-third of the activity measured using the portable air sampler (first column Table 15).

Airborne radioactivity measurements within the Furnace Building showed the best agreement between the two different air sampling system's results (Table 21 and first column of Table 16). Results from either sampling system showed fairly constant equilibrium concentrations within the decay chain series except for polonium-210 content, which appears to be elevated by up to an order of magnitude compared to the other decay chain members. This elevationis not solely relatable to the previously mentioned uncertainty in the analyses.

In summary, both the cascade impactor and the portable air sampler results show elevated airborne radioactivity concentrations, the most notable being polonium-210 and lead-210 for

general plant working areas as compared to the airborne concentrations measured in the Technical Building Library. The lowest airborne radioactivity concentration measured in a working area of the Thermal Process Plant was obtained in the Furnace Building. The highest airborne activity was measured in the Calciner.

Particle Size Analysis

Figures 2, 3, and 4 present log-normal probability plots of data from the cascade impactor sampling. Figures 2 and 3 present the data from the Calciner and Proportioning Building. The plots are based on the net concentrations (gross minus activity in blank filters) of radium-226 and polonium-210. The gross concentrations for thorium and uranium were used in Figure 2 because the blank concentrations were below the detection limits. The gross concentrations were high enough that the blank concentrations made a negligible contribution. Only radium-226 concentrations were used for Figures 3 and 4 because the contribution from the other radionuclides in all of the stages was not sufficiently above the concentrations in the blank filters to provide meaningful results.

The plotted points represent the percent of the collected activity (calculated in terms of pCi/m^3) less than the indicated sizes. The actual points were calculated by allocating all the activity in the stages and final filter subsequent to a stage to the specified size cut off for the stage (Elder et al., 1974). The manufacturer specified size cutoffs for the impactor at 40 cfm are 7.2, 3.0, 1.5, 0.95, and 0.49 micrometers (equivalent aerodynamic diameter). These are partially based on a study by Willeke (1975).

Thus, in the actual calculations, the percent of the total activity on the 3.0, 1.5, 0.95, and 0.49 and final filter stages was designated as less than 7.2 micrometers. The percent of the total activity on the final filter was ascribed to the 0.49

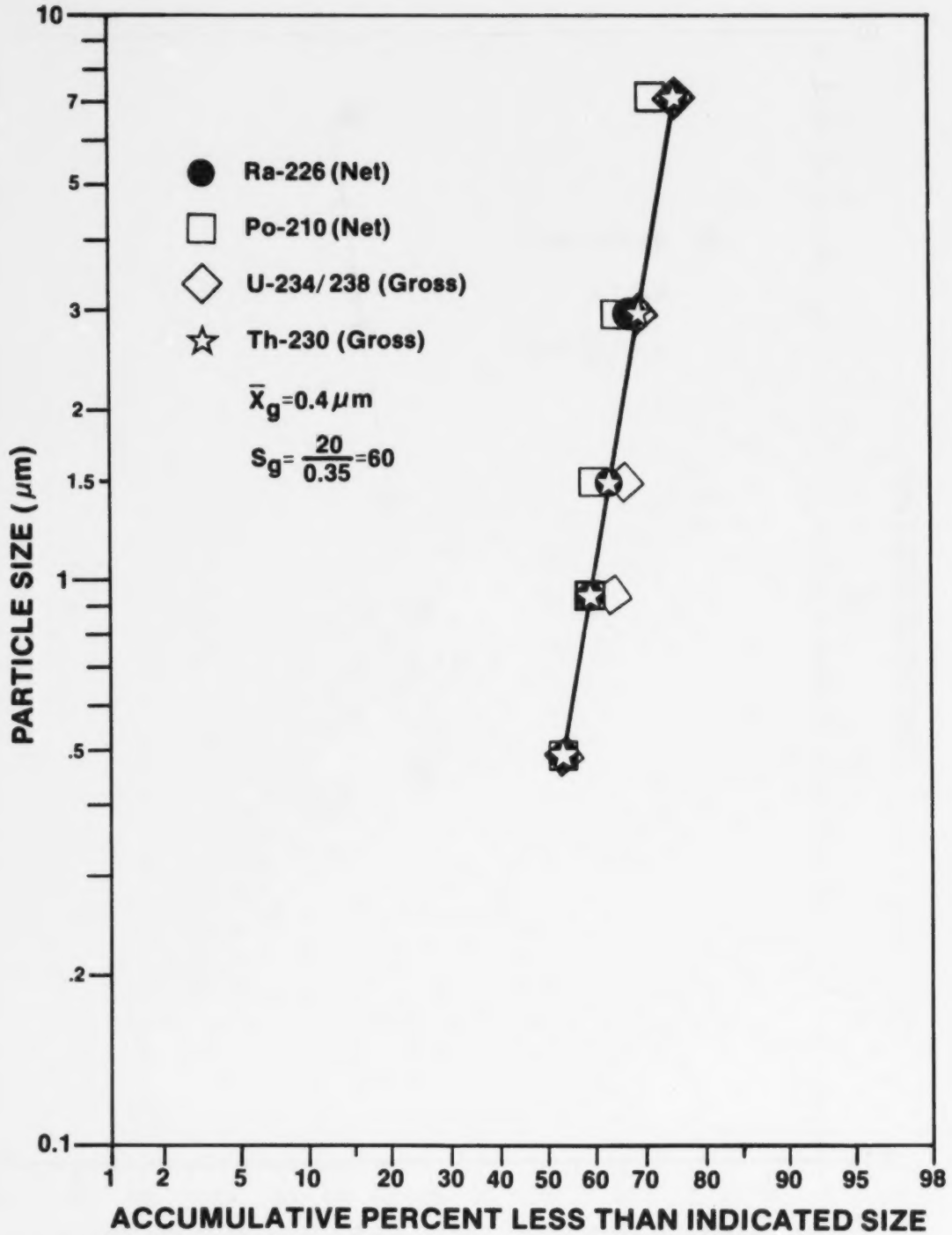

Figure 2. Particle size distribution Calciner (Log-probability plot of selected data taken from Table 18.)

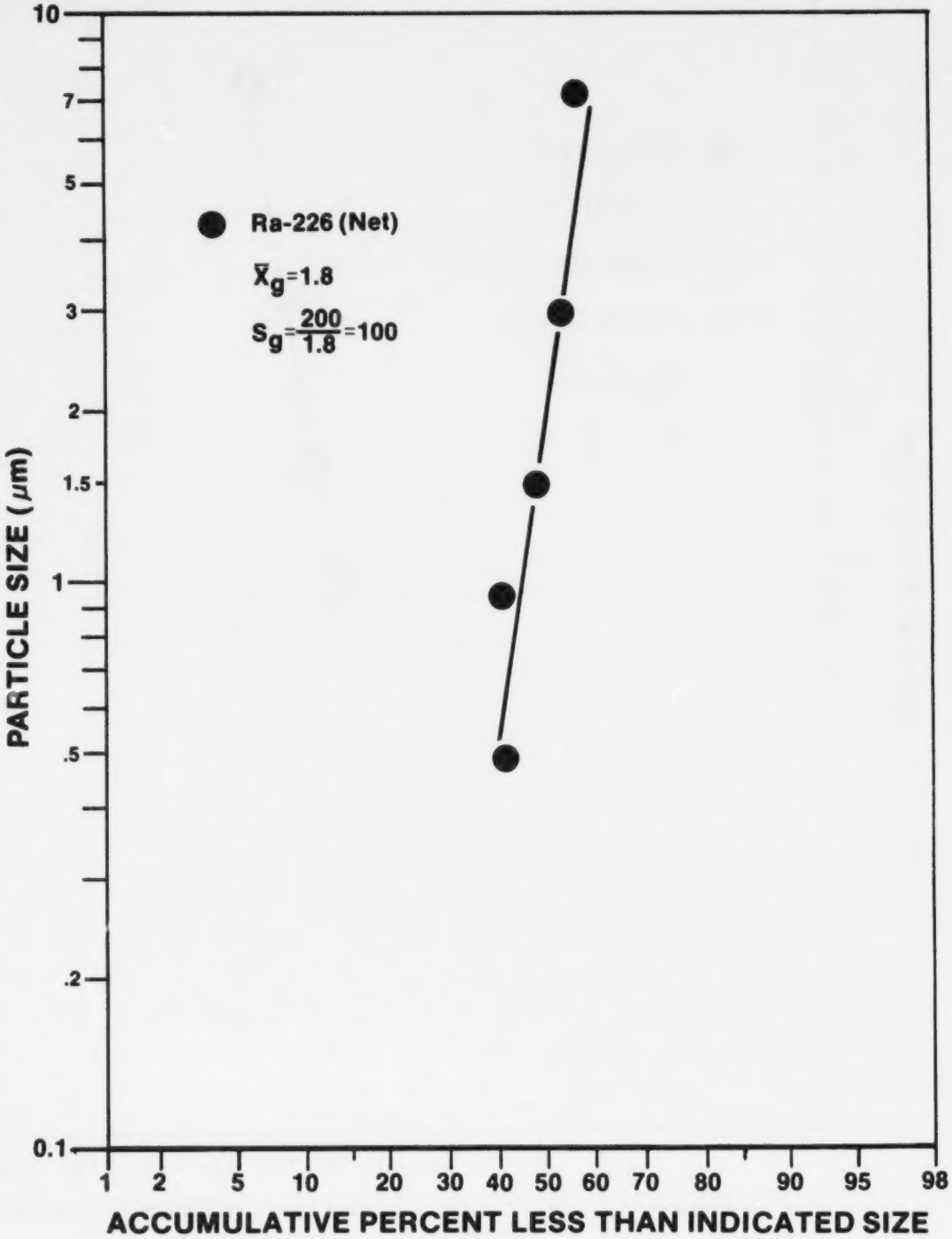

Figure 3. Particle size distribution Proportioning Building
(Log-probability plot of selected data taken from Table 19.)

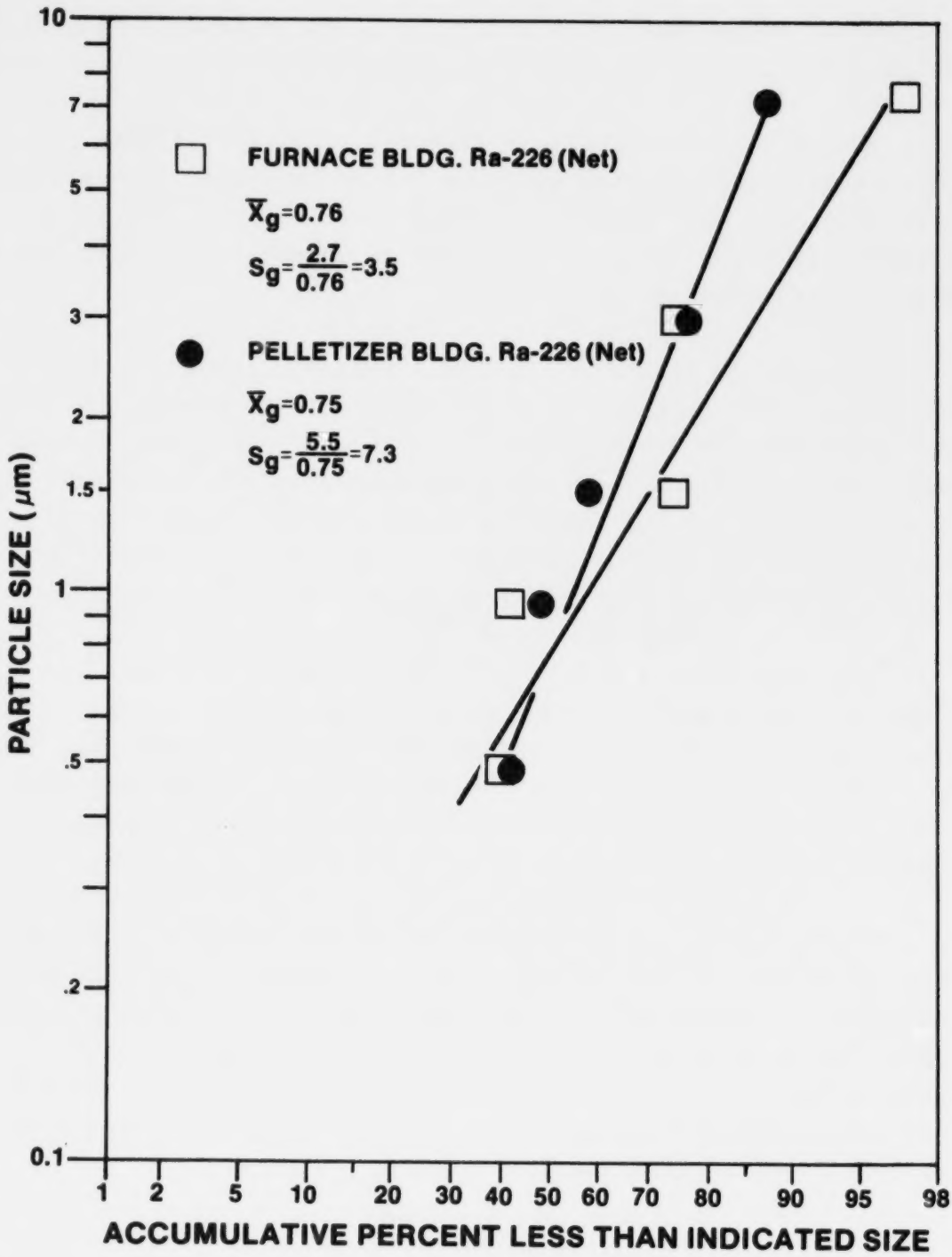

Figure 4. Particle size distribution (Log-probability plot of selected data taken from Tables 20 and 21.)

micrometer stage, etc. The lines are based on hand-drawn estimates and not based on regression analysis.

The plotted distributions (Figures 2, 3, and 4) have not been statistically tested to demonstrate the appropriateness of describing the data with a lognormal distribution. The log-probability plots are used as a tool to summarize the data and describe its central tendency.

The data in Figures 2, 3, and 4 indicate geometric means ($\overline{X}_g$) of one to three micrometers which are within expected values for atmospheric and industrial dusts (Lee, 1972; Elder et al., 1974; and Willeke, 1975). The geometric standard deviations (S_g) of 110 and 60, for the data in Figures 2 and 3, appear to be excessive, while those in Figure 4 (areas of lower concentrations) are within ranges seen for ambient dust (Lee, 1972).

The high geometric standard deviations for the data in Figures 2 and 3 probably reflect the composite of several size distributions. That is, there may be several particle size distributions from the plant in conjunction with the ambient dust. The Calciner (Figure 2) and Proportioning Building (Figure 3) results indicate concentrations above the ambient levels.

Knuth (1976), in his evaluation of the impactor, indicated that it tended to underestimate the size distribution and overestimate the geometric standard deviation. Thus, it is possible that this is also reflected in the data. Although, Knuth's results indicated that this appeared to be more of a problem for the larger size distributions ($\overline{X}_g$ of 7 micrometers) but was less of a problem for distributions around 4 micrometers.

Results from Lee (1972) indicate average geometric standard deviations of up to 20 for ambient air. Elder et al., (1974) indicate geometric standard deviations in the thirties for

plutonium processes. Thus, the high values noted in this study may be realistic. Furthermore, the data are not sufficient to break the indicated distribution into component parts which would have lower geometric standard deviations. The data in Figure 4 are for concentrations which appear to be near ambient levels.

STACK SAMPLING

Stack sampling was conducted in late September 1975 using the RAC Train Stacksamplr* and methods specified in the Federal Register, Volume 36, No. 247 (December 23, 1971). Representative samples were obtained from each process-type discharge stack; however, every plant discharge stack was not sampled (e.g., only one each of the Medusa stacks was sampled, but this one sample should be representative of the other remaining Medusa discharge stacks).

Particulate matter was collected on a glass fiber filter (2.5-inch diameter) which was subsequently analyzed for natural radioactivity content. To determine the blank filter natural radioactivity content, two sets of unused filters were analyzed and these results are shown in Table 25. Also shown is the extrapolated filter content based on the area ratio between the 2.5-inch and the 4-inch diameter filters of 0.391 times the 4-inch diameter activity content as reported by Eadie and Bernhardt, (1976). These results show fairly good agreement between the measured and the extrapolated activity contents. Both filters are made from the same type of material. There was less analytical uncertainty for the 4-inch filters because the sample was based on a larger amount of material. The extrapolated filter content (last column of Table 25) has been used as a blank filter activity content which has been subtracted from the gross analytical result to obtain the reported "net result". No blank

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* Research Appliance Corporation (RAC), Gibsonia, PA.

TABLE 25. RADIOACTIVE CONTENT OF BLANK GLASS FIBER FILTER (2.5-INCH DIAMETER)*
(pCi/filter ± two-sigma counting error term)

Radionuclide	5-Filter Composite	6-Filter Composite	Average Blank Filter Content+	Extrapolated Filter Content++
Ra-226	0.52 ± 0.14	0.42 ± 0.16	0.47 ± 0.21	0.14 ± 0.035
Po-210	0.29 ± 0.11	0.058 ± 0.015	0.17 ± 0.11	<0.066
U-234	0.060 ± 0.038	0.022 ± 0.018	<0.041	0.039 ± 0.012
U-235**	<0.0012	0.0012± 0.00079	<0.0012	0.0014± 0.00039
U-238	<0.026	0.025 ± 0.017	<0.026	0.031 ± 0.0078
Th-230	<0.075	0.089 ± 0.040	<0.082	0.078 ± 0.031
Th-232	<0.080	0.029 ± 0.023	<0.055	0.051 ± 0.0078
Pb-210	No data	<0.088	<0.088	<0.13
Ra-228	<0.81	No data	<0.81	<0.62

*Average 2.5-inch filter mass ± two standard deviations of 0.2159 ± 0.0050 grams.
**U-235 calculated based on natural U-235 to U-238 activity ratio of 1:21.45 (0.0466).
+Average of the 5-filter and 6-filter composite results.
++Extrapolated value based on area ratio between 2.5-inch and 4-inch diameter filters of 0.391 times the 4-inch diameter activity content (Eadie and Bernhardt, 1976).

subtractions have been made for the three radionuclides (radium-228, polonium-210, and lead-210) which are at the analytical minimum detectable activity (MDA) level. The net results for the stack effluent discharge radioactivity concentrations are reported in Table 26. If a polonium/lead-210 blank value of 0.1 pCi/filter is used, the blank activity for a 1 m^3 sample is 0.1 pCi/m^3. It is then evident that the measured values are significantly above this estimated blank value, except for the Coke Bag House.

MASS AND ACTIVITY BALANCE

General mass and activity balances are presented as accumulative bar graphs in Figure 5. These balances only include the major process streams (one percent of total mass or activity) and are limited to the solids input and output streams and stack effluents. Additional details concerning the balances and the parameters used in the calculations are given in Appendix D.

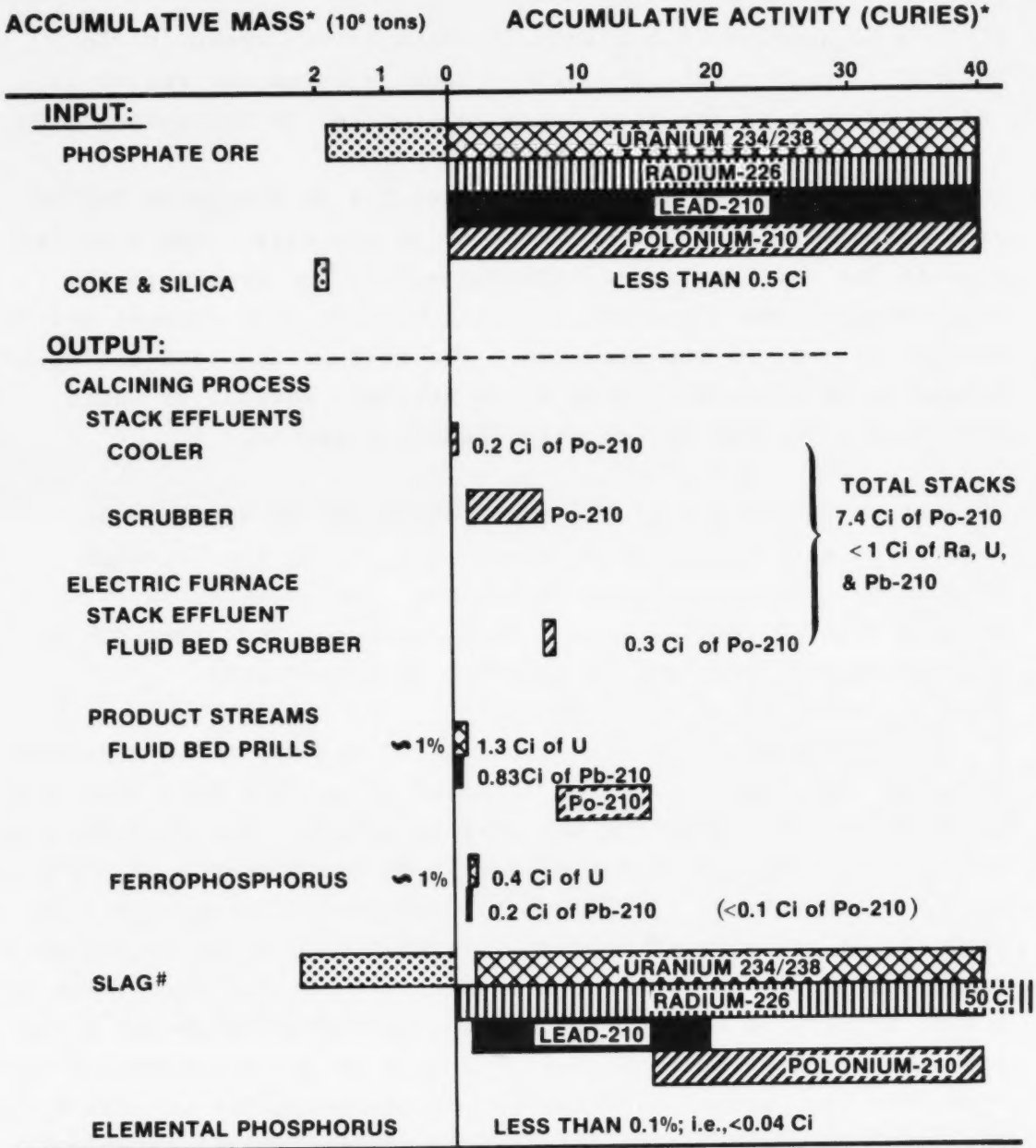
It is emphasized that the balances are based on a limited amount of data (roughly one-cubic meter stack discharge samples of selected stacks and limited grab samples of the product and by-product streams) taken at a point in time. There are also uncertainties in the results due to counting statistics and analytical variances, potential sample collection and treatment biases, and in the case of the lead-210 air results, actual errors in the analysis. The polonium-210 results for the stack samples are sufficiently greater than the lead-210 results so that the error in the lead-210 results has essentially no impact on the polonium-210 results (i.e., the ingrowth correction for polonium from lead is minimal).

The balances indicate that the phosphate ore contributes 90 percent of the input mass and essentially 100 percent of the input activity which is estimated to be 40 curies per year of uranium-234 and -238, radium-226, lead-210, and polonium-210

TABLE 26. STACK SAMPLING RESULTS
(NET RADIONUCLIDE CONCENTRATION ± TWO-SIGMA ERROR TERM IN pCi/m^3-DRY STANDARD CUBIC METERS)

Location Sampled Volume (date)	Ra-226	Po-210*	U-234	U-235	U-238	Th-230	Th-232	Pb-210*	Ra-228*
Reclaim No. D Bag House 1.09m^3 (9/22/75)	0.59 ± 0.22	17 ± 3.6	0.64 ± 0.15	<0.039	0.57 ± 0.14	0.81 ± 0.26	<0.16	2.7 ± 0.52	<1.5
2-2 Calciner Cooler 1.26m^3 (9/23/75)	1.7 ± 0.31	61 ± 6.3	15 ± 1.2	0.76 ± 0.16	13 ± 1.1	1.9 ± 0.33	0.36± 0.16	3.3 ± 0.55	<1.3
2-2 Calciner Scrubber 1.43m^3 (9/23/75)	0.31 ± 0.16	1700± 140	0.30 ± 0.10	<0.022	0.32 ± 0.092	0.55 ± 0.17	0.27± 0.12	1.0 ± 0.40	<0.00095
Coke Bag House 0.73m^3 (9/25/75)	0.38 ± 0.27	<2.8	<0.018	<0.0091	<0.026	<0.014	<0.011	3.9 ± 0.79	4.4 ± 2.5
Fluid Bed Scrubber 0.86m^3 (9/26/75)	1.2 ± 0.35	530 ± 45	0.20 ± 0.12	<0.070	0.14 ± 0.12	<0.019	0.20± 0.15	9.3 ± 1.6	<2.0
Medusa 4-West 1.12m^3 (9/30/75)	0.60 ± 0.23	25 ± 4.1	0.23 ± 0.10	<0.028	0.21 ± 0.10	<0.070	<0.018	8.2 ± 0.93	<1.5

*Gross analytical result.



* THE INPUT AND OUTPUT VALUES ARE INDICATED AS ACCUMULATIVE VALUES. THUS, SUBSEQUENT ENTRIES ARE ADVANCED ALONG THE SCALE TO INDICATE THE SUM-TOTAL FROM PREVIOUS ENTRIES. ACCUMULATIVE OUTPUT VALUES IN EXCESS OF THE INPUT VALUES REFLECT UNCERTAINTIES IN THE ANALYTICAL AND MASS FLOW DATA.

THE SLAG Po-210 VALUE IS BASED ON A LESS THAN VALUE OF 16 pCi/g (i.e., ALTHOUGH ANALYTICAL RESULTS INDICATE THE PRESENCE OF Po-210 IN THE SLAG, THE INDICATED MASS BALANCE VALUE HAS AN UNCERTAINTY APPROACHING 100 PERCENT——SEE APPENDICES C AND D).

Figure 5. Annual mass and activity balance of major streams - the thermal process plant

(this also applies to thorium-230 which is not shown in the balance). Essentially all of the input uranium and radium-226, and 90 percent of the mass are accounted for in the output slag.

Only 50 percent of the input lead-210 is accounted for in exit streams, almost all of which is in the slag. The lead-210 results for the slag are considered valid, but errors in the stack samples and fluid bed prills could possibly account for 10 percent or more of the lead-210. The rest of the lead-210 would appear to be associated with fluid streams, several of which indicated high lead-210 results (Tables 8 and 10).

About 20 percent of the input polonium-210 activity is associated with the stack effluents (primarily the Calciner scrubbers). The analytical uncertainty for this estimate is greater than 10 percent (i.e., counting error and error due to correcting for polonium-210 ingrowth from lead-210).

Over 50 percent of the polonium-210 appears to be associated with the slag, but this value is based on using a less than value of 16 pCi/g for estimating the slag inventory. The polonium-210 results in Table C-3 in Appendix C imply the presence of polonium in the slag, but the correction for polonium-210 ingrowth from lead-210 in the slag (Appendix A) introduces considerable uncertainty in these results. Furthermore, the assumed slag value (16 pCi/g) appears to be contradicted by the low polonium-210 value for the calcined rock. That is, if most of the polonium-210 is lost from the process material during calcining, it can not be in the final by-product slag (see Appendix D for actual inventory values). Thus, it appears that whereas the polonium-210 inventory value for calcined rock (4 Ci or less) is probably an underestimate, the estimate for polonium-210 in slag (26 Ci or less) may also be an overestimate. If the estimate for polonium-210 in slag is too high, then the output polonium-210 is not accounted for in the products used for the balance (i.e., use of 26 Ci in slag accounts for 100 percent).

SUMMARY

External gamma radiation measurements conducted in various working areas of the FMC Corporation's Thermal Process Plant near Pocatello, Idaho ranged from background to 100 μR/h. General work areas averaged about 42 mrem per work year (2000 hours), and the maximum net dose equivalent rate (directly on the slag pile) was estimated to be 182 mrem per work year. The dose received from natural background radiation in the Pocatello area was estimated to be about 79 mrem per year due to a cosmic and terrestrial background exposure rate of about 9 μR/h.

Ambient radon-222 concentrations in several buildings ranged from 0.17 to 1.4 pCi/l, except for one value of 11 pCi/l, which was measured in the Control Room of the Condenser and Fluid Bed Building. Ambient radon measured in two process control rooms were higher than would normally be expected from natural terrestrial background sources alone. Appreciable exposures from radon and its progeny may be experienced by personnel working in several indoor areas of the plant. Therefore, additional ambient radon or working level determinations should be conducted to better define this exposure source term.

Extensive radiochemical analyses of the input ore, calcined briquettes and the waste product slag, indicate natural radioactivity contents in excess of natural rocks and soils. Except for the lead-210 and polonium-210 activity, there appears to be little difference in the radioactivity content of these three materials. For the input ore, the various decay chain members are in secular equilibrium with the parent uranium-238. The calcined briquettes and slag showed depleted lead-210 and polonium-210 contents. The slag indicates somewhat less lead-210 than the calcined rock or briquettes (but statistically not different) indicating there is some additional loss of lead-210 in the electric furnace after the calcining.

The polonium-210 results for slag and calcined rock are contradictory in that the slag results are higher than the rock. Although the 100 percent error terms cause the results to overlap, it appears that the calcined rock results may be low. Also, the assumption that the slag concentration is equal to the less than value is probably an overestimate. In any case, these data in conjunction with the stack results, indicate the calcining process is the primary source for releasing airborne polonium-210.

A general mass and activity balance (Figure 5) indicates that the phosphate ore accounts for essentially all of the input activity, and the slag accounts for the output activity, except for lead-210 and polonium-210. Only about 50 percent of the input lead-210 is accounted for in the output solids streams (mostly in the slag) and stack effluents. Most of the remaining lead-210 is apparently in the liquid streams (e.g. scrubbers, etc). Up to about 50 percent of the input polonium-210 is accounted for in the slag and about 20 percent in the stack effluents.

The highest airborne particulate radioactivity was measured in the area of the south side discharge of the Calciner. These airborne concentrations are orders of magnitude greater than results obtained in the Technical Building Library, which may be considered as typical background levels for the purposes of this study.

Particle size characterization (of selected cascade impactor results) indicates geometric means of one to three micrometers, which are within the expected values for atmospheric and industrial dusts. The average of all cascade impactor sampling results indicates that roughly 50 percent of the arithmetic total airborne radioactivity is contained in the particle size fraction less than one micrometer.

Stack sampling of representative effluent discharge points showed radium-226 release concentrations ranging from 0.31 to 1.7 pCi/m^3, and total uranium ranged from less than 0.054 pCi/m^3 to about 29 pCi/m^3.

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APPENDIX-A

RADIOCHEMICAL ANALYTICAL METHODS

All analyses were completed at the Environmental Monitoring and Support Laboratory in Las Vegas, Nevada (EMSL-LV). The following sections present brief descriptions of the analytical methods employed for this study. Specific details of the procedures are contained in the Handbook of Radiochemical Analytical Methods, F. B. Johns, ed. (1975).

Analysis of Radium-226, Radium-228 and Lead-210

A sequential method for the determination of radium-226, radium-228, and lead-210 in environmental samples has been developed by the EMSL-LV. This method is initiated by the precipitation of radium from the sample aliquot using barium sulfate. Barium-radium-sulfate is then dissolved in a diethylenetriaminepentaacetate disodium solution and transferred to an emanation tube and the radon allowed to come to equilibrium, approximately 30 days ingrowth with its parent - radium. Radium-226 ($T_{\frac{1}{2}}$ = 1602 years) decays by alpha emission to radon-222 ($T_{\frac{1}{2}}$ = 3.8 days). Radon-222, a noble gas, is then collected from the liquid by a de-emanation technique. Radon-222 is usually counted for 30 minutes by alpha scintillation at four and one-half hours after the de-emanation step to allow for the build-up of the daughters.

The solution from the radium-226 determination is saved and the total radium is reprecipitated. Radium-228 ($T_{\frac{1}{2}}$ = 6.1 years) is a beta emitter and decays to actinium-228 ($T_{\frac{1}{2}}$ = 6.13 hours). The actinium is allowed to ingrow for three days and is extracted

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with diethylhexylphosphoric acid and back extracted with nitric acid. The actinium-228 is beta counted for 30 minutes in a low-level beta counter.

Lead is also precipitated with the radium sulfate in the original solution. Advantage is taken of the 30-day storage, for radon-222 ingrowth, to allow the bismuth-210 to grow in. Lead-210 ($T_{\frac{1}{2}}$ = 20.4 years) decays by beta emision to bismuth-210 ($T_{\frac{1}{2}}$ = 5.01 days). The bismuth-210 is precipitated from the supernatant liquid in the radium-228 separation step. Bismuth is converted to an oxide, dissolved in nitric acid, mounted on a two-inch planchet and beta counted for 30 minutes.

Analysis of Isotopic Uranium and Thorium

Samples are decomposed utilizing techniques of nitric-hydrofluoric acid digestion, potassium fluoride fusion or ignition. The residues are dissolved in dilute nitric acid and successive sodium and ammonium hydroxide precipitations are performed in the presence of boric acid to remove fluoride and soluble salts. The hydroxide precipitate is dissolved, the solution is adjusted to 9N in hydrochloric acid, and uranium is absorbed on an anion exchange column, separating it from thorium. Iron is removed from the column by washing with hydrochloric acid and the uranium is eluted with dilute hydrochloric acid. The thorium is converted to a nitrate form and absorbed on the same anion exchange column separating it from calcium and other interferences. The thorium is then eluted with 9N hydrochloric acid. The uranium is electrodeposited on stainless steel discs from an ammonium sulfate solution and subsequently counted by alpha spectrometry. Usually 1000-minute counting times are used for analysis. Chemical yields are normally determined by the recovery of internal tracer standards (e.g., uranium-232 and thorium-234) added at the beginning of the analysis.

Analysis of Polonium-210

Samples are decomposed by digestion with hydrofluoric acid and nitric acid in the presence of lead carrier and a polonium-208 tracer. Polonium is co-precipitated with lead sulfide from a dilute acid solution separating it from calcium, iron, and other interferences. The sulfide precipitate is dissolved in dilute hydrochloric acid and polonium is spontaneously deposited on a nickel disk. Polonium-210 and polonium-208 tracer are measured by alpha spectrometry. Usually, 1000-minute counting times are used for analysis.

Polonium-210 Activity Estimations

Due to the time delay between sample collection and radiochemical analysis for polonium-210, the following considerations have been utilized to estimate the polonium-210 activity in a sample.

1. Polonium-210 decays with a radiological half-life of 138 days. Therefore, a decay correction factor should be considered for samples which are held for a relatively long period between collection and analysis.

2. Concurrently, there will be some polonium-210 ingrowth from lead-210 and bismuth-210 contents of the original sample during the elapsed time between collection and analysis.

   The consideration of radioactive series transformations is discussed in detail in Kaplan (1958). For the specific case of lead-210, the decay scheme is as shown in Figure A-1. The solution of the system of differential equations, which describe this lead-210 decay series, was derived by Bateman and is summarized here.

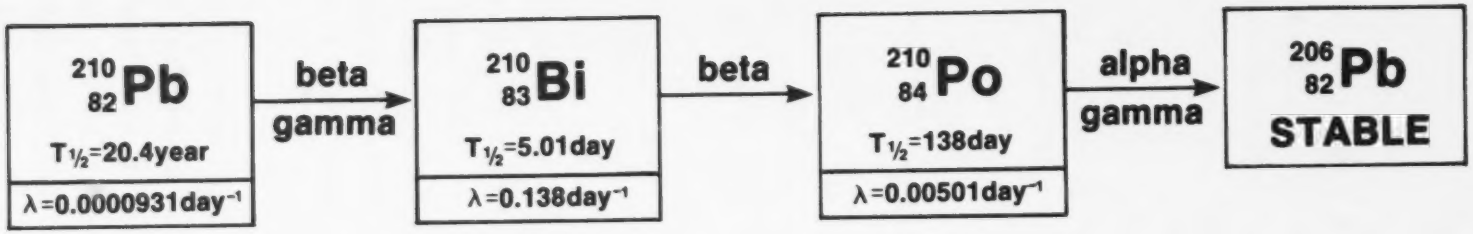

FIGURE A-1. LEAD-210 DECAY SCHEME

a. For polonium-210 ingrowth from lead-210 in the original sample:

$$A_{Po}^{Pb} =$$

$$1.01\ A_{Pb}^{o} e^{-0.0000931t} + 0.0377\ A_{Pb}^{o} e^{-0.138t} - 1.06\ A_{Pb}^{o} e^{-0.00501t}$$

Where

A_{Po}^{Pb} = polonium-210 activity at time of measurement due to ingrowth from lead-210 in sample

A_{Pb}^{o} = lead-210 activity at time of sample collection

t = Elapsed time (in days) between sample collection and analysis

Since the bismuth term is negligible and allowing for typical values of t, the following simplification of the above expression results:

$$A_{Po}^{Pb} = A_{Pb}^{o} (1.01 - 1.06\ e^{-0.00501t})$$

b. For polonium-210 ingrowth from bismuth-210 in the original sample:

$$A_{Po}^{Bi} = 0.0377\ A_{Bi}^{o} e^{-0.00501t} - 0.0377\ A_{Bi}^{o} e^{-0.138t}$$

Where

A_{Po}^{Bi} = polonium-210 activity at time of measurement due to ingrowth from bismuth-210 in sample.

A_{Bi}^{o} = bismuth-210 activity at time of sample collection

This term is insignificant for typical values of t and even in the case of the complete decay of bismuth-210, the polonium-210 ingrowth factor would be less than four percent (i.e., decay constant for polonium-210 divided by the decay constant for bismuth-210 equals 0.036).

c. The polonium-210 decay correction term:

$$A_{Po}^{Po} = A_{Po}^{o} e^{-0.00501t}$$

Where

A_{Po}^{Po} = polonium-210 activity at time of measurement due to decay of original polonium-210 in sample.

A_{Po}^{o} = Original polonium-210 activity in the sample at time of collection.

Therefore, the polonium-210 activity at the time of measurement (A_{Po}) is the summation of the two source terms (i.e., due to the ingrowth from lead-210 and due to the decay of the original polonium-210 in the sample).

$$A_{Po} = A_{Po}^{Pb} + A_{Po}^{Po}$$

or

$$A_{Po} = A_{Pb}^{o} (1.01 - 1.06\ e^{-0.00501t}) + A_{Po}^{o} e^{-0.00501t}$$

d. Due to the relatively long half-life of lead-210 ($T_{\frac{1}{2}}$ = 20.4 years), the lead-210 activity at the time of measurement (A_{Pb}) is essentially equivalent to the original Pb-210 activity in the sample (A_{Pb}^{o}). Solving the above expression for the original polonium-210 activity in the sample at the time of collection (A_{Po}^{o}) yields:

$$A_{Po}^{o} = e^{0.00501t} [A_{Po} - A_{Pb} (1.01 - 1.06\ e^{-0.00501t})]$$

All reported polonium-210 analytical results are calculated values obtained from the above equation. Whenever the apparent polonium-210 content from its ingrowth from lead-210 exceeds the measured polonium-210 activity, a non-detectable (ND) value has been reported.

e. The estimated counting error term associated with this calculated polonium-210 activity is simply the sum of squares of the individual counting error terms, or:

$$\sigma^2_{A_{Po}o} = (e^{0.005t})^2[\sigma^2_{A_{Po}} + \sigma^2_{A_{Pb}} (1 - 1.1\, e^{-0.005t})^2]$$

Where

$\sigma_{A_{Po}o}$ = Estimated counting error term of the calculated polonium-210 activity in the sample at the time of collection.

$\sigma_{A_{Po}}$ = Counting error term of the measured polonium-210 activity.

$\sigma_{A_{Pb}}$ = Counting error term of the measured lead-210 activity.

In the tables, the estimated counting error terms at the 95 percent confidence level (i.e., twice $\sigma_{A_{Po}o}$) have been included for the calculated polonium-210 activities.

Throughout this report, the symbol for less than (<) has been used to indicate the equivalence of the error term (at the 95 percent confidence level) to the reported result.

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TABLE B-1. AIRBORNE PARTICULATE SAMPLING IN THE CALCINER AREA
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.

Radionuclide	South Side Discharge,* Calciner Belt #1 Dump Area - 1218 to 1500 hrs. (5/5/75)		No. 1 Control Room - ** 1027 to 1520 hrs. (5/5/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	2.7 ± 0.18	38 ± 2.4	0.019 ± 0.010	24 ± 13
Po-210	66 ± 4.9	560 ± 70	<0.0078	<88
U-234	1.5 ± 0.12	20 ± 1.7	0.0091 ± 0.0023	12 ± 3.1
U-235***	0.044 ± 0.014	0.62 ± 0.20	(<0.00032)	(<0.43)
U-238	1.5 ± 0.12	21 ± 1.7	0.0068 ± 0.0023	9.2 ± 3.1
Th-230	2.8 ± 0.15	39 ± 2.1	0.0091 ± 0.0045	12 ± 6.1
Th-232	0.055 ± 0.022	0.77 ± 0.30	0.0045 ± 0.0023	6.1 ± 3.1
Pb-210	0.40 ± 0.28	120 ± 9.5	0.17 ± 0.066	210 ± 83
Ra-228	<0.23	<3.2	<0.085	<120

*Sampled Volume of 7.9m^3, with a dust particulate load of 0.5663g.
**Sampled Volume of 17.6m^3, with a dust particulate load of 0.0131g.
***U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).

TABLE B-2. AIRBORNE PARTICULATE SAMPLING IN THE PROPORTIONING BUILDING
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.

Radionuclide	Lower Level, 1st Pillar* West of Shack, 0947 to 1309 hrs. (5/6/75)		Lower Level Control Room** 0944 to 1316 hrs. (5/6/75)		Upper Level, Top West+ 1140 to 1315 hrs. (5/6/75)		Upper Level Control Room++ 0930 to 1411 hrs. (5/6/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.37 ± 0.045	15 ± 1.8	0.13 ± 0.027	17 ± 3.5	2.9 ± 0.21	16 ± 1.2	0.49 ± 0.054	30 ± 3.4
Po-210	5.3 ± 0.89	210 ± 36	<2.7	350 ± 71	<3.4	<21	3.8 ± 0.94	240 ± 60
U-234	0.30 ± 0.036	12 ± 1.5	0.097 ± 0.017	13 ± 2.2	2.8 ± 0.23	16 ± 1.3	0.29 ± 0.034	18 ± 2.1
U-235	0.014 ± 0.0069	0.54 ± 0.27	0.0045 ± 0.0039	0.59 ± 0.51	0.13 ± 0.027	0.71 ± 0.15	0.012 ± 0.0058	0.73 ± 0.35
U-238	0.32 ± 0.038	13 ± 1.5	0.11 ± 0.019	15 ± 2.4	2.8 ± 0.23	16 ± 1.3	0.32 ± 0.035	19 ± 2.2
Th-230	0.72 ± 0.062	29 ± 2.5	0.21 ± 0.034	27 ± 4.3	4.9 ± 0.25	27 ± 1.4	0.50 ± 0.048	31 ± 3.0
Th-232	0.037 ± 0.014	1.5 ± 0.58	<0.0028	<0.35	0.12 ± 0.036	0.66 ± 0.20	0.015 ± 0.0083	0.91 ± 0.51
Pb-210	0.63 ± 0.11	25 ± 4.5	0.24 ± 0.097	30 ± 12	6.6 ± 0.48	37 ± 2.7	0.74 ± 0.10	45 ± 6.5
Ra-228	<0.15	<6.0	<0.12	<15	<0.39	<2.2	<0.11	<7.0

*Sampled Volume of 12.4m^3, with a dust particulate load of 0.3109g.
**Sampled Volume of 12.5m^3, with a dust particulate load of 0.0978g.
+Sampled Volume of 4.44m^3, with a dust particulate load of 0.7853g.
++Sampled Volume of 14.7m^3, with a dust particulate load of 0.2387g.

TABLE B-3. AIRBORNE PARTICULATE SAMPLING IN THE PELLETIZER BUILDING
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.*

Radionuclide	Operator Level, Middle** Lower Level, 1000 to 1537 hrs. (5/5/75)		Upper Level, Top East+ 1003 to 1530 hrs. (5/5/75)		Upper Control Room ++ 1006 to 1528 hrs. (5/5/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.28 ± 0.033	20 ± 2.3	0.41 ± 0.044	24 ± 2.5	0.17 ± 0.03	29 ± 4.9
Po-210	1.1 ± 0.046	82 ± 31	1.6 ± 0.042	86 ± 24	ND	ND
U-234	0.45 ± 0.042	32 ± 2.9	0.26 ± 0.029	15 ± 1.7	0.10 ± 0.015	17 ± 2.6
U-235	0.023 ± 0.0096	1.6 ± 0.68	0.014 ± 0.0058	0.80 ± 0.33	0.0052 ± 0.0034	0.86 ± 0.57
U-238	0.24 ± 0.03	17 ± 2.1	0.25 ± 0.028	14 ± 1.6	0.10 ± 0.016	17 ± 2.6
Th-230	0.50 ± 0.044	35 ± 3.1	0.53 ± 0.05	30 ± 2.9	0.19 ± 0.028	32 ± 4.7
Th-232	0.019 ± 0.0089	1.3 ± 0.63	0.025 ± 0.011	1.5 ± 0.66	0.014 ± 0.0079	2.2 ± 1.3
Pb-210	0.66 ± 0.10	46 ± 7.1	0.60 ± 0.092	37 ± 5.7	0.30 ± 0.07	50 ± 12
Ra-228	<0.11	<7.7	<0.10	<6.0	<0.095	<16

*ND indicates non-detectable.
**Sampled Volume of 18.7m^3, with a dust particulate load of 0.2660g.
+Sampled Volume of 14.3m^3, with a dust particulate load of 0.2502g.
++Sampled Volume of 18.0m^3, with a dust particulate load of 0.1083g.

TABLE B-4. AIRBORNE PARTICULATE SAMPLING IN THE FURNANCE BUILDING
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.*

Radionuclide	Control Room Level** Between Furnace 3 & 4 0906 to 1606 hrs.(5/7/75)		Tapping Level+ Between Furnace 3 & 4 0910 to 1607 hrs.(5/7/75)		Control Room for ++ Furnace 3 & 4, Behind Panel 0853 to 0851 hrs. (5/7 to 6/75)	
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g
Ra-226	0.047 ± 0.012	17 ± 4.4	0.014 ± 0.0062	8.8 ± 4.0	0.0091 ± 0.003	12 ± 4.0
Po-210	0.63 ± 0.23	230 ± 84	<0.048	<69	0.16 ± 0.057	210 ± 79
U-234	0.011 ± 0.0041	3.8 ± 1.5	0.0095 ± 0.0046	6.2 ± 3.0	0.0024 ± 0.0012	3.2 ± 1.6
U-235*	<0.00097	<0.35	(0.00051)	(0.34)	(0.0002)	(0.26)
U-238	0.011 ± 0.0041	3.8 ± 1.5	0.011 ± 0.0052	7.2 ± 3.4	0.0042 ± 0.0016	5.5 ± 2.1
Th-230	0.02 ± 0.0079	7.0 ± 2.8	0.0089 ± 0.0053	5.4 ± 3.4	0.0036 ± 0.0017	4.8 ± 2.6
Th-232	0.0066 ± 0.0047	2.4 ± 1.7	0.0042 ± 0.0036	2.7 ± 2.2	<0.00089	<1.2
Pb-210	0.27 ± 0.054	98 ± 20	0.25 ± 0.055	150 ± 34	0.027 ± 0.013	36 ± 18
Ra-228	<0.069	<25	<0.067	<44	<0.019	<25

*U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).
**Sampled Volume of 24.3m^3, with a dust particulate load of 0.0678g.
+Sampled Volume of 23.4m^3, with a dust particulate load of 0.0370g.
++Sampled Volume of 84m^3, with a dust particulate load of 0.0623g.

TABLE B-5. AIRBORNE PARTICULATE SAMPLING
GROSS RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM.*

Radionuclide	Technical Building ** Library- 0935 (5/5/75) to 1421 hrs.(5/6/75)		Condensor and Fluid Bed *** Control Room- 0904 to 1444 hrs (5/6/75)		Old Kiln Building (Maintenance) + Open Area - East End 0912 to 1412 hrs. (5/6/75)		P-4 Loading Area ++ 0933 to 1438 hrs.(5/8/75)
	pCi/m^3	pCi/g	pCi/m^3	pCi/g	pCi/m^3	pCi/g	High Vol. Sampler pCi/m^3
Ra-226	0.0013 ± 0.0012	21 ± 19	0.012 ± 0.008	9.6 ± 6.4	0.27 ± 0.034	24 ± 3.0	0.0032 ± 0.0008
Po-210	ND	ND	0.29 ± 0.25	250 ± 210	43 ± 4.9	3900 ± 430	0.038 ± 0.027
U-234	0.00093 ± 0.00062	15 ± 9.9	0.015 ± 0.0038	13 ± 3.1	0.20 ± 0.025	18 ± 2.1	0.0028 ± 0.00058
U-235*	(0.000056)	(0.93)	(0.00051)	(0.44)	0.0079 ± 0.0044	0.69 ± 0.38	<0.00014
U-238	0.0012 ± 0.00062	20 ± 9.9	0.011 ± 0.0038	9.4 ± 3.1	0.21 ± 0.025	18 ± 2.2	0.0033 ± 0.00065
Th-230	0.0019 ± 0.00062	30 ± 9.9	0.017 ± 0.0057	14 ± 4.7	0.33 ± 0.037	29 ± 3.3	0.0047 ± 0.00022
Th-232	<0.00031	<4.9	0.0057 ± 0.0038	4.7 ± 3.1	0.024 ± 0.011	2.1 ± 0.95	0.00053 ± 0.00022
Pb-210	0.019 ± 0.0086	320 ± 140	0.20 ± 0.059	160 ± 47	0.58 ± 0.089	51 ± 7.9	0.018 ± 0.0038
Ra-228	<0.011	<190	<0.081	<66	<0.092	<8.1	<0.0045

*U-235 results reported in parenthesis are calculated based on U-235 to U-238 natural activity ratio of 1 to 21.45 (0.0466).
ND indicates non-detectable.
**Sampled Volume of 130m^3, with a dust particulate load of 0.0081g.
***Sampled Volume of 21m^3, with a dust particulate load of 0.0256g.
+Sampled Volume of 16.4m^3, with a dust particulate load of 0.1862g.
++Sampled Volume of 346m^3.

APPENDIX C

ADDITIONAL SAMPLING OF INPUT AND SLAG PRODUCTS

APPENDIX C

ADDITIONAL SAMPLING RESULTS

On December 23, 1976, additional samples of the input ore, calcined briquettes and of the waste product slag were obtained from the FMC Corporation. Six samples of each type of material were analyzed for natural radioactivity content at both the Environmental Monitoring and Support Laboratory (EMSL) in Las Vegas, Nevada and at the Eastern Environmental Radiation Facility in Montgomery, Alabama. These results are shown in Tables C-1, C-2, and C-3.

Average analytical results except for lead-210, are well within the 95 percent confidence level, indicating excellent analytical agreement between the two laboratories. Considering the statistical fluctuations in the analytical results, there appears to be little difference, except for lead and polonium-210, in the radioactivity content of the input ore, calcined briquettes and slag. However all three materials exhibit elevated levels of natural radioactivity as compared to natural rocks and soil, as discussed in the main text.

As noted in the text and in the footnotes for Tables C-1, C-2, and C-3 the EMSL lead-210 results are in error. Thus, these results, although reported, are discounted in determining the averages.

For the input ore samples, the various decay chain members are in secular equilibrium with the parent uranium-238. The calcined briquettes and slag showed depleted polonium-210 contents, averaging roughly one-half of the activity of the other individual decay chain members. Therefore, it appears that the calcining process is the mechanism for generating the elevated airborne concentrations of polonium-210 as reported in the main text.

TABLE C-1. FMC ORE (PHOSPHATE ROCK) SAMPLES (SAMPLED 12/23/76) RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, in pCi/g

	SAMPLE #1		SAMPLE #2		SAMPLE #3		SAMPLE #4	
	EERF*	EMSL**	EERF	EMSL	EERF	EMSL	EERF	EMSL
Ra-266	24.3 ± 0.243	26 ± 0.93	24.0 ± 0.24	27 ± 0.94	21.3 ± 0.213	14 ± 0.68	19.4 ± 0.194	30 ± 1.0
Po-210*	21.5 ± 3.2	24 ± 2.4	22.8 ± 3.3	22 ± 1.3	25.2 ± 3.7	21 ± 1.7	19.4 ± 1.5	22 ± 1.7
U-234	22.0 ± 3.30	21 ± 5.2	20.6 ± 2.99	22 ± 2.4	21.4 ± 2.99	23 ± 2.3	21.7 ± 3.15	27 ± 4.7
U-235	1.02 ± 0.204	<0.78	1.12 ± 0.217	0.98 ± 0.41	0.985 ± 0.187	0.89 ± 0.37	0.978 ± 0.191	<0.45
U-238	22.3 ± 3.35	22 ± 5.5	21.0 ± 3.05	20 ± 2.2	22.2 ± 3.11	23 ± 2.3	21.8 ± 3.15	26 ± 4.7
Th-232	0.483 ± 0.0676	0.30 ± 0.11	0.516 ± 0.0722	0.37 ± 0.13	0.439 ± 0.0636	0.49 ± 0.16	0.441 ± 0.0639	0.42 ± 0.14
Th-230	21.4 ± 0.641	23 ± 1.1	24.1 ± 0.723	24 ± 1.0	22.2 ± 0.666	19 ± 0.96	22.8 ± 0.648	23 ± 0.99
Th-228	0.415 ± 0.0622	-------	0.484 ± 0.0677	--------	0.427 ± 0.064	---------	0.479 ± 0.0646	----------
Th-227	0.857 ± 0.128	-------	0.723 ± 0.123	--------	0.713 ± 0.118	---------	0.857 ± 0.124	----------
Pb-210	27.7 ± 0.9	2.7 ± 0.69	31.8 ± 1.9	6.4 ± 0.83	29.4 ± 1.5	4.7 ± 0.76	27.7 ± 1.0	5.8 ± 0.8
Ra-228	<1.0	0.78 ± 0.76	<1.0	0.74 ± 0.66	<1.0	0.98 ± 0.68	<1.0	0.67 ± 0.65

*EERF - Eastern Environmental Radiation Facility, Montgomery, Alabama.

**EMSL - Environmental Monitoring and Support Laboratory, Las Vegas, Nevada; blanks indicate no data.

*Since these values are for geological material that has not been processed (i.e., calcined), it is assumed the polonium and lead-210 values are in equilibrium as are the other radionuclides; therefore, these polonium values have not been corrected for ingrowth from lead. EMSL lead-210 results have been found to be in error by up to a factor of five too low, as discussed in the text.

SAMPLE #5		SAMPLE #6		AVERAGE ± TWO STANDARD DEVIATIONS in pCi/g	
EERF	EMSL	EERF	EMSL	EERF	EMSL
21.1 ± 0.211	29 ± 0.98	20.8 ± 0.208	53 ± 1.3	21.8 ± 3.86	30 ± 25
21.5 ± 1.5	23 ± 1.6	22.4 ± 0.9	21 ± 2.4	22.1 ± 3.82	22 ± 2.3
21.8 ± 3.05	24 ± 3.9	20.7 ± 3.00	21 ± 3.0	21.4 ± 1.18	23 ± 4.6
1.17 ± 0.221	1.2 ± 0.78	0.876 ± 0.175	0.99 ± 0.56	1.02 ± 0.212	0.88 ± 0.51
21.9 ± 3.07	24 ± 4.0	20.5 ± 2.97	22 ± 3.2	21.6 ± 1.43	23 ± 4.1
0.454 ± 0.0635	0.46 ± 0.15	0.479 ± 0.0646	0.36 ± 0.13	0.469 ± 0.0594	0.40 ± 0.14
22.1 ± 0.662	23 ± 1.0	22.0 ± 0.66	17 ± 0.87	22.4 ± 1.86	22 ± 5.6
0.435 ± 0.063	----------	0.449 ± 0.0629	--------	0.448 ± 0.0563	----------
0.820 ± 0.127	----------	0.685 ± 0.113	--------	0.776 ± 0.155	----------
15.4 ± 1.3	6.5 ± 0.82	31.9 ± 1.8	7.2 ± 1.0	27.3 ± 12.3	5.6 ± 3.3
< 1.0	< 0.65	<1.0	<0.81	<1.0	<0.77

TABLE C-2. RMC CALCINED BRIQUETTES (SAMPLED 12/23/76) RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, in pCi/g

	SAMPLE #1		SAMPLE #2		SAMPLE #3		SAMPLE #4	
	EERF*	EMSL**	EERF	EMSL	EERF	EMSL	EERF	EMSL
Ra-226	24.2 ± 0.242	28 ± 0.97	21.8 ± 0.218	20 ± 0.82	22.4 ± 0.224	31 ± 1.0	20.3 ± 0.203	25 ± 0.91
Po-210*	(7.1 ± 0.6)	(9.2 ± 1.1)	(6.1 ± 0.6)	(8.5 ± 0.88)	(7.4 ± 0.6)	(11 ± 1.2)	(8.1 ± 0.9)	(9.3 ± 0.86)
Corrected**	ND	----------	ND	----------	ND	----------	ND	----------
U-234	23.8 ± 3.56	22 ± 2.0	22.3 ± 3.12	22 ± 1.7	21.9 ± 3.18	22 ± 2.0	23.2 ± 3.25	23 ± 1.5
U-235	1.16 ± 0.225	1.1 ± 0.48	0.975 ± 0.185	0.94 ± 0.30	0.861 ± 0.168	1.0 ± 0.33	1.14 ± 0.216	0.94 ± 0.26
U-238	23.5 ± 3.53	24 ± 2.0	23.0 ± 3.22	22 ± 1.8	22.3 ± 3.23	24 ± 1.9	23.6 ± 3.30	23 ± 1.6
Th-232	0.479 ± 0.0574	0.50 ± 0.15	0.404 ± 0.0606	0.52 ± 0.15	0.575 ± 0.0719	0.44 ± 0.14	0.410 ± 0.0636	0.49 ± 0.14
Th-230	21.6 ± 0.648	23 ± 1.0	23.0 ± 0.689	23 ± 0.98	22.7 ± 0.68	23 ± 1.0	23.8 ± 0.713	23 ± 0.95
Th-228	0.435 ± 0.0543	----------	0.520 ± 0.065	----------	0.493 ± 0.0641	----------	0.518 ± 0.0699	----------
Th-227	0.694 ± 0.104	----------	0.700 ± 0.108	----------	0.918 ± 0.133	----------	0.982 ± 0.137	----------
Pb-210*	19.1 ± 0.3	8.1 ± 0.90	20.5 ± 0.3	6.5 ± 0.85	25.1 ± 0.6	8.7 ± 0.93	17.4 ± 0.6	5.2 ± 0.77
Ra-228	<1.0	1.3 ± 0.84	<1.0	<0.73	<1.0	1.5 ± 0.83	<1.0	< 0.74

*EERF - Eastern Environmental Radiation Facility, Montgomery, Alabama.

**EMSL - Environmental Monitoring and Support Laboratory, Las Vegas, Nevada

*The EMSL lead-210 results have been found to be in error by up to a factor of five too low, as discussed in the text. The polonium-210 results in parenthesis represent values at date of analysis; the other polonium-210 values have been corrected for ingrowth from lead-210 as described in Appendix A.

**ND indicates non-detectable; blanks indicate no data.

SAMPLE #5		SAMPLE #6		AVERAGE ± TWO STANDARD DEVIATIONS in pCi/g	
EERF	EMSL	EERF	EMSL	EERF	EMSL
20.7 ± 0.207	29 ± 0.98	21.6 ± 0.216	32 ± 1.0	21.8 ± 2.77	28 ± 8.8
(8.2 ± 1.0) <2.65	(11 ± 1.1) --------	(7.9 ± 0.9) ND	(9.0 ± 0.89) ----------	(7.47 ± 1.58) <2.65	(9.7 ± 2.1) --------
23.1 ± 3.23	26 ± 2.5	23.0 ± 3.56	23 ± 2.3	22.9 ± 1.36	23 ± 3.1
0.870 ± 0.178	1.2 ± 0.42	1.10 ± 0.237	1.1 ± 0.43	1.02 ± 0.269	1.0 ± 0.21
23.7 ± 3.32	25 ± 2.4	23.9 ± 3.7	25 ± 2.5	23.3 ± 1.18	24 ± 2.3
0.465 ± 0.0674	0.42 ± 0.14	0.522 ± 0.0705	0.42 ± 0.13	0.476 ± 0.131	0.47 ± 0.087
21.4 ± 0.641	24 ± 1.0	20.9 ± 0.627	24 ± 1.0	22.2 ± 2.22	23 ± 1.0
0.406 ± 0.0649	---------	0.491 ± 0.0687	---------	0.477 ± 0.0929	---------
0.877 ± 0.136	---------	0.691 ± 0.117	---------	0.810 ± 0.261	---------
12.8 ± 1.3	5.0 ± 0.77	15.2 ± 1.5	9.7 ± 0.94	18.4 ± 8.60	7.2 ± 3.9
<1.0	< 0.78	<1.0	<0.63	<1.0	< 0.95

TABLE C-3. FMC SLAG (SAMPLED 12/23/76) RADIONUCLIDE CONCENTRATION ± TWO-SIGMA COUNTING ERROR TERM, in pCi/g

	SAMPLE #1		SAMPLE # 2		SAMPLE #3		SAMPLE # 4	
	EERF*	EMSL**	EERF	EMSL	EERF	EMSL	EERF	EMSL
Ra - 226	22.8 ± 0.228	37 ± 1.1	23.2 ± 0.232	40 ± 1.2	24.8 ± 0.248	36 ± 1.1	32.5 ± 0.325	37 ± 1.1
Po - 210+	(9.8 ± 1.0)	(8.3 ± 1.3)	(2.2 ± 0.2)	(2.4 ± 0.50)	(12.6 ± 2.1)	(11 ± 1.1)	(3.7 ± 0.5)	(3.3 ± 0.45)
Corrected++	8.27± 2.86	---	ND	---	23.7 ± 5.85	---	ND	---
U - 234	28.4 ± 4.25	27 ± 4.4	21.2 ± 2.96	27 ± 2.4	21.6 ± 2.91	27 ± 2.5	27.2 ± 3.81	29 ± 2.4
U - 235	1.59 ± 0.326	< 0.63	1.00 ± 0.19	1.0 ± 0.37	1.09 ± 0.201	1.1 ± 0.41	1.29 ± 0.245	0.95 ± 0.35
U - 238	29.4 ± 4.41	25 ± 4.2	22.2 ± 3.1	26 ± 2.4	21.6 ± 2.92	29 ± 2.6	27.7 ± 3.88	25 ± 2.3
Th - 232	0.648 ± 0.081	0.52 ± 0.19	0.632 ± 0.0789	0.20 ± 0.055	0.533 ± 0.0693	0.47 ± 0.15	0.672 ± 0.084	0.66 ± 0.16
Th - 230	25.5 ± 0.764	29 ± 1.4	25.8 ± 0.774	11 ± 0.40	26.3 ± 0.789	28 ± 1.1	26.6 ± 0.797	23 ± 0.95
Th - 228	0.617 ± 0.0771	---	0.713 ± 0.0784	---	0.648 ± 0.0778	---	0.696 ± 0.0835	---
Th - 227	0.981 ± 0.142	---	0.993 ± 0.139	---	0.974 ± 0.136	---	0.762 ± 0.118	---
Pb - 210+	11.6 ± 1.0	2.3 ± 0.67	11.1 ± 1.7	0.86 ± 0.61	6.6 ± 0.6	2.1 ± 0.68	7.8 ± 0.7	0.69 ± 0.62
Ra - 228	<1.0	< 0.80	< 1.0	< 0.76	< 1.0	1.6 ± 0.86	< 1.0	< 0.76

*EERF - Eastern Environmental Radiation Facility, Montgomery, Alabama.

**EMSL - Environmental Monitoring and Support Laboratory, Las Vegas, Nevada

+The EMSL lead-210 results have been found to be in error by up to a factor of five too low, as discussed in the text. The polonium-210 results in parenthesis represent values at date of analysis; the other polonium-210 values have been corrected for ingrowth from lead-210 as described in Appendix A.

++ND indicates non-detectable; blanks indicate no data.

SAMPLE #5		SAMPLE #6		AVERAGE ± TWO STANDARD DEVIATIONS in pCi/g	
EERF	EMSL	EERF	EMSL	EERF	EMSL
32.3 ± 0.323	37 ± 1.1	33.3 ± 0.333	23 ± 0.87	28.2 ± 10.1	35 ± 12
(8.9 ± 0.7) ND	(11 ± 0.95)	(8.2 ± 1.3) ND	(8.0 ± 0.72)	(7.57± 7.82) <16.0	(7.3± 7.4)
19.7 ± 2.46	24 ± 3.3	20.4 ± 2.65	27 ± 2.0	23.1 ± 7.46	27 ± 3.2
1.63 ± 0.293	0.87 ± 0.51	0.931 ± 0.168	1.2 ± 0.33	1.26 ± 0.601	0.96 ± 0.40
18.6 ± 2.32	24 ± 3.3	20.3 ± 2.64	28 ± 2.0	23.3 ± 8.57	26 ± 3.9
0.627 ± 0.0783	0.73 ± 0.22	0.683 ± 0.0854	0.70 ± 0.18	0.632 ± 0.107	0.55 ± 0.40
28.1 ± 0.843	33 ± 1.5	27.3 ± 0.819	31 ± 1.2	26.6 ± 1.93	26 ± 16
0.672 ± 0.0839	--------	0.775 ± 0.093	----------	0.687 ± 0.110	---------
1.03 ± 0.144	--------	1.07 ± 0.144	----------	0.963 ± 0.215	---------
----------	1.4 ± 0.66	16.7 ± 2.8	4.2 ± 0.82	10.8 ± 7.88	<1.9
<1.0	<0.80	<1.0	<0.76	<1.0	<0.91

APPENDIX D

MASS AND ACTIVITY BALANCE

The data in Table D-1 show the parameters and calculated values for the mass and activity balances. The field study was organized to try and sample the primary plant streams, with emphasis on effluent streams. Thus, the data for a complete mass balance (i.e., internal plant liquid streams) is not available. Furthermore, all the data obtained has not been utilized directly in the balance. In some cases, mass flow information was not available and in other instances, it was apparent that a given stream did not contribute a significant amount of activity (i.e., less than one percent of the total). The present mass balance is limited to data on solids and stack effluents. That is, liquid effluent streams have not been included in the balances.

The primary input mass and activity is due to the phosphate ore. Since the radionuclide values are statistically the same and there is reasonable cause to assume equilibrium, the values have been averaged and the input is taken as 40 Ci per year for each radionuclide.

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APPENDIX D - TABLE D-1. MASS AND ACTIVITY BALANCE

MATERIAL	RADIONUCLIDE CONC. (pCi/Gram or m^3)[a]	MATERIAL FLOW[b,c]	MATERIAL FLOW PER YEAR Gram or m^3/year	ACTIVITY CURIES[d]
INPUT				
Phosphate Ore	(pCi/gram)	1.8E6 Ton/yr	1.6E12 g/yr	
U	22 ± 3			36 ± 14%
Ra-226	26 ± 19			42 ± 73%
Pb-210	27 ± 12			44 ± 44%
Po-210	22 ± 3			36 ± 14%
			Average ± 1 standard deviation	40 ± 4Ci
Coke	(pCi/g)	0.19E6 Ton/yr	0.17E12 g/yr	
U	<0.5 ± 0.5			0.086± 100%
Ra-226	0.78± 0.17			0.13 ± 22%
Pb-210	2.4 ± 0.6			0.41 ± 25%
Po-210	<1.3 ± 1.3			0.22 ± 100%
Silica	(pCi/g)	0.125E6 Ton/yr	0.11E12 g/yr	
U	1 ± 1			0.1 ± 100%
Ra-226	1.7 ± 0.24			0.2 ± 14%
Pb-210	0.67± 0.55			0.08 ± 82%
Po-210	2.6 ± 0.9			0.3 ± 35%
IN PROCESS				
Calcined Rock	(pCi/g)	1.5E6 Ton/yr	1.4E12 g/yr	
U	24 ± 2			33 ± 8%
Ra-226	25 ± 8.6			34 ± 34%
Pb-210	18 ± 8.6			24 ± 48%
Po-210	<2.7 ± 2.7			<4 ± 100%
OUTPUT				
Calcining Process Stack Effluents				
Calciner Cooler	(pCi/m^3)	100,700 cfm x $\frac{0.517 \text{ Dry Std. cfm}^f}{\text{stack cfm}}$ x (4 coolers[e])	3.10E9 m^3/yr	
U	14 ± 1			0.04 ± 7%
Ra-226	1.7 ± 0.3			0.005 ± 18%
Pb-210	3.2 ± 0.6			0.01 ± 19%
Po-210	61 ± 6			0.2 ± 10%
Calciner Scrubber	(pCi/m^3)	106,900 cfm x $\frac{0.636 \text{ Dry Std. cfm}^f}{\text{stack cfm}}$ x (4 scrubbers[e])	4.05E9 m^3/yr	
U	0.31± 0.10			0.001 ± 32%
Ra-226	0.31± 0.16			0.001 ± 52%
Pb-210	0.86± 0.5			0.003 ± 58%
Po-210	1700 ±140			6.9 ± 8%

Electric Furnace
Stack Effluent

Fluid Bed Scrubber	(pCi/m^3)	68,700 cfm x $\frac{0.581 \text{ Dry Std. cfm}}{\text{stack cfm}}$[f] x (1 scrubber[e])	5.94E8 m^3/yr	
U	0.17± 0.12			0.0001±71%
Ra-226	1.2 ± 0.35			0.0007±29%
Pb-210	9.2 ± 1.7			0.005 ±18%
Po-210	530 ±45			0.31 ± 8%
Products & By-products				
Fluid Bed Prills	(pCi/g)	18,000 Ton/yr	0.016E12 g/yr	
U	<81 ± 81			1.3 ±100%
Ra-226	13 ± 1			0.21 ± 8%
Pb-210	52 ± 2			0.83 ± 4%
Po-210	440 ± 27			7.0 ± 6%
Ferrophosphorus	(pCi/g)	22,000 Ton/yr	0.02E12 g/yr	
U	20 ± 5			0.4 ± 25%
Ra-226	0.27± 0.11			0.005± 41%
Pb-210	1.1 ± 0.6			0.02 ± 55%
Po-210	<0.6 ± 0.6			0.01 ±100%
Slag	(pCi/g)	1.8E6 Ton/yr	1.6E12 g/yr	
U	25 ± 7			40 ± 28%
Ra-226	32 ± 13			51 ± 41%
Pb-210	11 ± 8			18 ± 73%
Po-210	<16 ± 16			26 ± 100%
Elemental Phosphorus	(pCi/g)	0.125E6 Ton/yr	0.11E12 g/yr	
Ra-226	0.021± 0.020			0.002± 95%
Pb-210	0.21 ± 0.17			0.02 ± 81%

FOOTNOTES:

a. Values based on Tables 3, 4, and 26. The net lead and polonium-210 values are based on 0.1 pCi of activity per filter (see Table 25). The U results are an average of the uranium-234 and -238 results. Less than values have been assumed to be equal to the less than value. Volumes are dry standard cubic meters.

b. Mass flow data based on private correspondence with the company and data obtained with the stack samples.

c. Exponential notation is used. That is, E12 means 10^{12}.

d. The percent error terms are based on the error terms for the concentration of activity in the material.

e. This factor is for the number of similar unsampled stacks at the plant.

f. Correction factors for the effluent at stack conditions are based on temperature, pressure, and moisture measurements made during the stack sampling.

APPENDIX E

UNCERTAINTIES IN LEAD-210 AND POLONIUM-210 AIR SAMPLE DATA

Tables E-1 and E-2 present a tabulation of the polonium-210 concentrations, as of time of analysis, and lead-210 concentrations, as of the time of collection, for air samples listed in Tables 13 to 21 of the text. This tabulation can be used to assess the uncertainty associated with individual lead-210 results and to estimate the correct value.

Comparison of the lead-210 interlaboratory duplicate results reported in Appendix C and special method evaluation samples indicates that the EMSL lead-210 results for solids and air samples may be underestimated by up to a factor of five or more. Due to the delay of about 280 days between sample collection and analysis for polonium-210, the polonium-210 values are generally indicative of the lead-210 collected in the sample. That is, while the polonium-210 collected in the sample would have decayed to 25 percent of its level at time of collection prior to analysis, the ingrowth of polonium-210 from lead-210 collected in the sample would have reached about 75 percent of the lead-210 value. Thus, the polonium-210 at time of analysis would be indicative of the lead-210 (within 50 percent) under the range of extreme conditions of a polonium-210:lead-210 ratio of one-tenth (normal ambient, NCRP, 1975) to three.

Therefore, the true lead-210 concentrations should fall between the reported lead-210 values and the polonium-210 values at time of collection (Tables E-1 and E-2). Furthermore, with minor exceptions, the lead-210 values should be near the high side of the range indicated.

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The notable exceptions are cases where the polonium-210 concentrations are excessively above the reported lead-210 values (about an order of magnitude). Specifically, the samples from the Calciner South Side Discharge (from Table 13) and the Old Kiln Building (from Table 17) indicate that the polonium-210: lead-210 ratio was probably greater than three and thus, especially for the Calciner sample, it appears the polonium-210 in these samples at time of analysis was primarily due to polonium-210 collection, and not due to ingrowth. For these samples the best estimate for lead-210 is about 2 pCi/m^3 or less (up to about five times the reported lead value).

For these two samples, and for a few other samples with a polonium-210:lead-210 ratio at time of analysis of five or more, there is good evidence of elevated polonium-210 and that the decay and ingrowth corrected values reported in Tables 13 to 21 of the text are correct. For the remaining air sample results, although there is some evidence of elevated polonium-210, the individual sampling results are insufficient to clearly denote even the positive presence of polonium-210 (beyond that ingrown from the lead-210). That is, a polonium-210:lead-210 ratio of up to four at time of analysis for these data (if analysis is at 280 days after collection) can be solely due to a factor of five error in the lead-210 analysis and 75 percent ingrowth of polonium-210 from the lead. Although such a situation of elevated lead-210 with no polonium-210 is not likely, much of the data are insufficient to demonstrate that it could not occur.

In summary, the best estimates for the lead-210 values are the polonium-210 values at time of analysis, except for the values denoted above. The polonium-210 values can only be approximated as discussed above.

APPENDIX E

TABLE E-1. ESTIMATED UNCERTAINTY FOR LEAD-210 IN AIR FILTERS

Referenced Table/ Location Description	Reported Lead-210 (pCi/m^3)	Polonium-210 at Time of Analysis (pCi/m^3)
Table 13-Calciner Area		
South Side Discharge	0.40 ± 0.28	17 ± 1.2
No. 1 Control Room	0.17 ± 0.066	0.13 ± 0.026
Table 14-Proportioning Bldg		
Lower Level	0.63 ± 0.11	1.8 ± 0.21
Lower Level Control Room	0.24 ± 0.097	0.85 ± 0.12
Upper Level	6.6 ± 0.48	5.8 ± 1.4
Upper Level Control Room	0.74 ± 0.10	1.5 ± 0.22
Table 15-Pelletizer Bldg		
Operator Level	0.66 ± 0.10	0.78 ± 0.079
Upper Level	0.60 ± 0.092	0.85 ± 0.073
Upper Control Room	0.30 ± 0.07	0.19 ± 0.028
Table 16-Furnace Building		
Control Room Level	0.27 ± 0.054	0.36 ± 0.042
Tapping Level	0.25 ± 0.055	0.20 ± 0.033
Control Room	0.027± 0.013	0.059 ± 0.01
Table 17		
Library	0.019± 0.0086	0.011 ± 0.0035
Condenser & Fluid Bed Control Room	0.20 ± 0.059	0.22 ± 0.034
Old Kiln Building	0.58 ± 0.089	10 ± 1.1
P-4 Loading Area	0.018± 0.0038	0.023 ± 0.0062

* These data are gross results prior to blank filter activity subtraction.

+ The statistical uncertainty of values as expressed by the two-sigma counting error must be considered in comparing values. For example 5 ± 2 is not statistically greater than 3 ± 1. Furthermore, counting error does not include other analytical and sample collection uncertainties.

APPENDIX E

TABLE E-2. ESTIMATED UNCERTAINTY FOR LEAD-210 IN AIR FILTERS*+

Referenced Table/ Location/Description	Reported Lead-210 (pCi/m^3)	Polonium-210 at Time of Analysis (pCi/m^3)
Table 18-Calciner Area		
Filter 1	0.57 ± 0.057	3.1 ± 0.26
2	0.28 ± 0.056	0.97 ± 0.089
3	0.14 ± 0.041	0.51 ± 0.057
4	0.13 ± 0.038	0.37 ± 0.052
5	0.065 ± 0.036	0.45 ± 0.056
Final Filter	1.1 ± 0.071	6.1 ± 0.43
Table 19-Proportioning Bldg		
Filter 1	0.33 ± 0.082	0.39 ± 0.066
2	0.18 ± 0.076	0.25 ± 0.12
3	0.11 ± 0.073	<0.11
4	0.14 ± 0.075	<0.41
5	0.11 ± 0.074	0.15 ± 0.051
Final Filter	0.49 ± 0.086	0.53 ± 0.067
Table 20-Pelletizer Bldg		
Filter 1	0.073 ± 0.027	0.045 ± 0.012
2	0.025 ± 0.017	0.020 ± 0.0073
3	0.058 ± 0.026	0.019 ± 0.009
4	<0.018	0.031 ± 0.012
5	0.049 ± 0.027	0.035 ± 0.025
Final Filter	0.088 ± 0.026	0.10 ± 0.014
Table 21-Furnace Building		
Filter 1	0.03 ± 0.011	0.021 ± 0.004
2	0.034 ± 0.011	0.021 ± 0.0043
3	0.027 ± 0.011	0.021 ± 0.0042
4	0.039 ± 0.012	0.037 ± 0.0052
5	0.064 ± 0.013	0.06 ± 0.0084
Final Filter	0.061 ± 0.013	0.076 ± 0.0098

* These data are gross results prior to blank filter activity subtraction.

\+ The statistical uncertainty of values as expressed by the two-sigma counting error must be considered in comparing values. For example 5 ± 2 is not statistically greater than 3 ± 1. Furthermore, counting error does not include other analytical and sample collection uncertainties.

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